GRAND GULF NUCLEAR STATION
INITIAL LICENSE CLASS LOT-309
NRC RO/SRO EXAMS

EXAM KEY

Question	Answer		Question	Answer		Question	Answer
1	B		34	A		67	B
2	A		35	D		68	B
3	A		36	D		69	D
4	B		37	A		70	D
5	D		38	B		71	A
6	D		39	A		72	A
7	D		40	B		73	B
8	B		41	C		74	C
9	C		42	C		75	A
10	B		43	C		76	B
11	B		44	D		77	C
12	A		45	B		78	D
13	A		46	D		79	C
14	D		47	C		80	C
15	C		48	D		81	B
16	D		49	C		82	C
17	A		50	B		83	B
18	B		51	A		84	B
19	A		52	A		85	C
20	B		53	C		86	D
21	D		54	A		87	B
22	C		55	A		88	A
23	D		56	C		89	B
24	A		57	D		90	A
25	C		58	C		91	B
26	A		59	A		92	D
27	D		60	B		93	B
28	D		61	C		94	B
29	A		62	D		95	D
30	A		63	A		96	B
31	B		64	D		97	A
32	B		65	A		98	B
33	B		66	D		99	B
						100	C

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor the Neutron monitoring system as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295001 AA1.06
	Rating	3.3

Question 1

The plant conditions are as follows:

- Reactor power is 70%
- The Flow Control Valves for Recirculation Loops A and B are open to their 70% position

The Flow Control Valve for Recirculation Loop B fails closed.

The resulting APRM UPSCALE rod block setpoints can be obtained at the:

A. Jet Pump P619 panel

B. APRM recorders on the P680 panel

C. FCTR flowcards on the P669-P672 panels

D. PBDS displays on the P669 and P670 panels

Answer: B

Explanation:

A. This is incorrect because the jet pump differential pressure transmitters do not serve the function of input to the flow-biased rod block (APRM UPSCALE) setpoints.

B. This is correct because each APRM recorder's "PUSH TO REC" pushbutton display's the flow biased rod block (APRM UPSCALE) setpoint.

C. This is incorrect because the FCTR flow cards provide no visible indication of the flow biased rod block setpoints.

D. This is incorrect because PBDS measures LPRM input only, and does not interface with recirculation flow input/output at all.

Technical References:

GLP-OPS-B3300, GLP-OPS-C5104, GLP-OPS-C5106, GFIG-OPS-B3300, ARI P680-5A-B10

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of Load shedding as it applies to PARTIAL OR COMPLETE LOSS OF A.C. POWER (CFR: 41.8 to 41.10).	Tier #	1
	Group #	1
	K/A #	295003 AK1.02
	Rating	3.1

Question 2

Following a Loss of Offsite Power, the Load Shedding and Sequencing System will:

A. Isolate PSW to Control Room air conditioning and ESF switchgear room coolers.

B. Isolate SSW to Control Room air conditioning and ESF switchgear room coolers.

C. Isolate PCW to the RWCU NRHX.

D. Align PSW to P51/P52/P53.

Answer: A

Explanation:

A. This is correct because loss of offsite power does cause this load shedding action per GFIG-OPS-R2100.

B. This is incorrect because SSW is actually aligned to provide service to these coolers in the event of this situation.

C. This is incorrect because CCW actually provides the cooling to the RWCU NRHX.

D. This is incorrect because SSW is the cooling supply to these panels that is manually aligned in this situation.

Technical References:

GLP-OPS-R2100, GFIG-OPS-R2100, FSAR Table 8.3-9

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8 to 55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret the extent of partial or complete loss of D.C. power as it applies to PARTIAL OR COMPLETE LOSS OF D.C. POWER (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295004 AA2.02
	Rating	3.5

Question 3

The plant is in MODE 1 with the Division II EDG in MAINTENANCE mode.

Following a loss of ESF Transformer 21, which of the following DC buses will no longer have power to its battery chargers?

A. 11DL

B. 11DH

C. 11DD

D. 11DC

Answer: A

Explanation:

A. This is correct because the 16AB bus will be deenergized as part of LSSS for Division II (BUV), and power will not be restored with the diesel generator.

B. This is incorrect because the 24 VDC chargers are supplied by ESF Division I.

C. This is incorrect because bus 15AA is not deenergized by LSSS for Div I ESF loads (being fed from ESF XMFR 11 instead).

D. This is incorrect because the Div III DG starts and energized the chargers on this bus within 2-3 seconds (73% BUV).

Technical References:

GLP-OPS-R2100, GLP-OPS-L1100, GFIG-OPS-R2100, GFIG-OPS-L1100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of Pressure effects on reactor level as it applies to MAIN TURBINE GENERATOR TRIP (CFR: 41.8 to 41.10).	Tier #	1
	Group #	1
	K/A #	295005 AK1.03
	Rating	3.5

Question 4

The reactor is at rated power at the beginning of the fuel cycle (BOC).

Following a main turbine trip, reactor water level would shrink:

A. until HPCS/RCIC automatically initiate to restore reactor level.

B. until feedwater control stabilizes level below the original level.

C. and swell repeatedly due to SRV cycling until feedwater control stabilizes level below the original level.

D. at a rate that requires low pressure means (LPCI/LPCS) to restore level.

Answer: B

Explanation:

A. This is incorrect because a main turbine trip would not result in a level decrease to the extent of initiation of HPCS or RCIC systems. This is characteristic for Loss of Vacuum or MSIV Closure events.

B. This is correct because the level will shrink due to the reactor pressure rise resulting from the turbine trip, as well as due to the loss of void production and core flow sweeping out the pre-scram voids. Feedwater restores level at a level below the original level as determined by setpoint setdown.

C. This is incorrect because this is indicative of a main turbine trip without bypass capability.

D. This is incorrect because even in the worst case pressure increase transient event (MSIV closure), HPCS/RCIC are shown to be able to provide water to restore reactor vessel level.

Technical References:

GLP-OPS-MCD12, FSAR Section 15.2.3, FSAR Table 15.2-5

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.8 to 55.41.10	

Examination Outline Cross-Reference	Level	RO
SCRAM - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13).	Tier #	1
	Group #	1
	K/A #	295006 Generic - 2.1.7
	Rating	4.4

Question 5

Following a reactor scram due to a loss of both reactor recirc pumps, operators have determined the following:

- Reactor steam dome temperature is 450^0F
- RWCU bottom head suction line temperature is 370^0F
- Recirculation Loop A discharge temperature is 405^0F
- Recirculation Loop B discharge temperature is 395^0F

According to procedure 05-1-02-I-1, Reactor Scram, a reactor cooldown:

A. Should be initiated because the difference between the reactor steam dome and the reactor bottom head drain line temperatures exceeds 50^0F

B. Is not needed because the difference between the reactor bottom head drain line and Recirculation Loops A or B temperatures does not exceed 50^0F

C. Is not needed because the difference between the reactor steam dome and the reactor bottom head drain line temperatures does not exceed 100^0F

D. Should be initiated because the difference between the reactor steam dome and Recirculation Loop B temperatures exceeds 50^0F

Answer: D

Explanation:

A. This is incorrect because the temperature differential between the reactor steam dome and the reactor bottom head drain line needs to exceed 100 F to require a reactor cooldown following the scram.

B. This is incorrect because there is no specific temperature differential requirement between the reactor bottom head drain line and the recirculation loops associated with reactor cooldown initiation following a scram.

C. This is incorrect because although the temperature differential between the reactor

steam dome and the reactor bottom head drain line does not exceed 100^0F (a reactor cooldown requirement following a scram), the situation in Answer D requires a reactor cooldown.

D. This is correct because the temperature differential exceeds one of the criteria for initiating a reactor cooldown following a scram per Procedure 05-1-02-I-1, Section 3.9.3.

Technical References:

Procedure 05-1-02-I-1, “Reactor Scram”

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.43.5, 55.45.12, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Control Room Abandonment - Knowledge of limiting conditions for operations and safety limits (CFR: 41.5 / 43.2 / 45.2).	Tier #	1
	Group #	1
	K/A #	295016 General-2.2.22
	Rating	4.0

Question 6

According to Technical Specifications, which of the following is required to be OPERABLE in MODES 1 and 2 ONLY?

A. High Pressure Core Spray

B. Primary Containment

C. Secondary Containment

D. Remote Shutdown System

Answer: D

Explanation:

A. This is incorrect because HPCS is required to be operable in modes 1, 2, and 3.

B. This is incorrect because PC is required to be operable in modes 1, 2, and 3.

C. This is incorrect because secondary containment is required to be operable in modes 1, 2, and 3.

D. This is correct per Technical Specification 3.3.3.2.

Technical References:

Technical Specifications 3.3.3.2, 3.5.1, 3.6.1.1, and 3.6.4.1, GLP-OPS-C6100, ONEP 10-S-01-11

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.43.2, 55.45.2	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and system loads (CFR: 41.7 / 45.8).	Tier #	1
	Group #	1
	K/A #	295018 AK2.01
	Rating	3.3

Question 7

A loss of the Component Cooling Water (CCW) system would result in the loss of cooling to the:

A. CRD pump seals.

B. Offgas vault refrigeration units.

C. Heater drain pump coolers.

D. RWCU non-regenerative heat exchanger.

Answer: D

Explanation:

A. This is incorrect because the CCW system cools the CRD pump oil coolers, but not the pump seals.

B. This is incorrect because this is a cooling load for the TBCW system.

C. This is incorrect because this is a cooling load for the TBCW system.

D. This is correct because it is one of the cooling loads for the CCW system listed in both GLP-OPS-P4200 and ONEP 05-1-02-V-1.

Technical References:

GLP-OPS-P4200, GLP-OPS-P4300, ONEP 05-1-02-V-1

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
PARTIAL OR TOTAL LOSS OF INST. AIR - Ability to manage the control room crew during plant transients (CFR: 41.10 / 43.5 / 45.12 / 45.13).	Tier #	1
	Group #	1
	K/A #	295019 Generic - 2.1.6
	Rating	3.8

Question 8

The control room crew observes the following plant indications:

- Reactor power is 95%
- INST AIR RCVR PRESS LO alarm on Panel P870 has annunciated
- Control Rod 03-21 has drifted from position 42 to 38
- All other control rods are in their expected positions

According to procedure 05-1-02-IV-1, Control Rod/Drive Malfunctions, the operating crew should:

A. Scram the reactor.

B. Fully insert control rod 03-21.

C. Restore rod 03-21 to position 42.

D. Verify the scram air header pressure regulating valve is operating correctly.

Answer: B

Explanation:

A. This is incorrect because a scram is not required unless there is more than one rod drifting.

B. This is correct because per Procedure 05-1-02-V-9, after the immediate action to find and attempt to restore air pressure, it directs the user to Procedure 05-1-02-IV-1 if there are any control rod drifts. In this procedure, if there is one rod drifting, the direction is to manually insert the one rod (Section 2.2 of procedure).

C. This is incorrect because the action is to drive the rod so it is fully inserted.

D. This is incorrect because although the rod is drifting as the result of low scram air header pressure, the problem is upstream of the pressure regulator as indicated by the

low air pressure alarm. The input for this alarm is sensed upstream of the scram air header pressure regulating valve.

Technical References:

Procedure 05-1-02-V-9, “Loss of Instrument Air;” Procedure 05-1-02-IV-1, “Control Rod/Drive Malfunctions;” GLP-OPS-P5300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor Alternate heat removal methods as they apply to LOSS OF SHUTDOWN COOLING (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295021 AA1.04
	Rating	3.7

Question 9

RHR ‘B’ is operating in Shutdown Cooling through the common suction.

No Recirc Pumps are running.

Per the Shutdown Cooling SOI, which of the following identifies the parameter to be used for monitoring and recording reactor coolant temperature?

A. Recirc loop suction temperature

B. RPV bottom head drain temperature

C. RHR ‘B’ Heat Exchanger inlet temperature

D. RHR ‘B’ Heat Exchanger outlet temperature

Answer: C

Explanation:

Per the E12-2 SOI, P/L 3.8.16.a, only ‘C’ is correct.

A, B, and D are incorrect; see explanation for ‘C’.

Technical References:

Procedure 04-1-01-E12-2, Shutdown Cooling and Alternate Decay Heat Removal Operation

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for responses of interlocks associated with fuel handling equipment as they apply to REFUELING ACCIDENTS (CFR: 41.5 / 45.6).	Tier #	1
	Group #	1
	K/A #	295023 EK3.02
	Rating	3.4

Question 10

For fuel handling operations, interlocks that prevent adding excessive reactivity to the core that could lead to a refueling accident are the:

A. Fuel handling platform bridge and trolley interlocks

B. Refueling platform bridge and main hoist interlocks

C. Fuel handling platform main hoist interlocks

D. Refueling platform frame-mounted and monorail-mounted auxiliary hoist interlocks

Answer: B

Explanation:

A. This is incorrect based on the Explanation for Answer B. Reactivity control is not one of the reasons stated for this interlock set.

B. This is correct because this is stated as one of the purposes for this set of interlocks in GLP-RF-F1101.

C. This is incorrect based on the Explanation for Answer B. Reactivity control is not one of the reasons stated for this interlock set.

D. This is incorrect based on the Explanation for Answer B. Reactivity control is not one of the reasons stated for this interlock set.

Technical References:

GLP-RF-F1101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor the emergency generators as they apply to HIGH DRYWELL PRESSURE (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295024 EA1.06
	Rating	3.7

Question 11

Following a transient, plant conditions are as follows:

- Reactor power is 0%
- Drywell pressure is 1.40 psig
- Reactor vessel level is +42”

Which of the following would result in a trip of a running Div I or Div II EDG?

A. Generator current of 1580 amps

B. Generator speed of 520 rpm

C. Lube oil temperature of 220^0F

D. Generator frequency of 59 Hz

Answer: B

Explanation:

A. This is incorrect because it is not a diesel generator parameter that would result in a trip in the given situation (normally would at 1575 A). See Explanation for Answer B.

B. This is correct because when a LOCA signal is present (Drywell pressure greater than 1.39 psig), only two trips are active for the diesel generators (speed ›517.5 rpm and generator differential overcurrent ›75 A). Other trip parameters are not in effect in this situation. A diesel generator speed of 520 rpm would result in a trip.

C. This is incorrect because it is not a diesel generator parameter that would result in a trip in the given situation (normally would at 200^0F). See Explanation for Answer B.

D. This is incorrect because it is not a diesel generator parameter that would result in a trip in the given situation (normally would at 59.5 Hz). See Explanation for Answer B.

Technical References:

GLP-OPS-P7500, GLP-OPS-MCD16, GLP-OPS-C7100, FSAR Figure 6.3-14

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between HIGH REACTOR PRESSURE and the reactor/turbine pressure regulating system (Plant Specific) (CFR: 41.7 / 45.8) (EK2.08 - RO)	Tier #	1
	Group #	1
	K/A #	295025 EK2.08
	Rating	3.7

Question 12

The plant is operating at rated power when one inboard MSIV fails closed.

Before operators take any action, how should the Bypass Control Valves automatically respond?

A. Remain closed.

B. Only the 'A' valve begins to open.

C. Only the 'A' and 'B' valves begin to open.

D. All three valves begin to open.

Answer: A

Explanation:

A single MSL isolation at rated power does cause a short-term reactor pressure rise which causes the HP Control Valves to open farther. The remaining three MSLs easily and quickly pick up the steam load of the now isolated single MSL, causing the HP Control Valves to return to their pre-event positions. The normally-closed BCVs should not have to open. Only 'A' is correct.

Technical References:

GLP-OPS-N3202, EHC Control Oil lesson plan

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of conservative decision making practices as it applies to Suppression Pool High Water Temp. (CFR: 41.10 / 43.5 / 45.12) (2.1.39 – RO)	Tier #	1
	Group #	1
	K/A #	295026 G2.1.39
	Rating	3.7

Question 13

The plant is operating at rated power.

Operators are preparing to place RCIC in service for its quarterly surveillance.

The Actions of Tech Spec 3.6.2.1 (Suppression Pool Average Temperature) allow a suppression pool temperature as high as 105°F before requiring that the RCIC test be suspended.

However, the RCIC surveillance procedure (06-OP-1E51-Q-0003) takes a more conservative approach to limiting the rise in suppression pool temperature.

Which of the following describes how the RCIC surveillance procedure addresses the rising suppression pool temperature?

Directs operators to…

A. initiate suppression pool cooling before suppression pool temperature reaches 95°F.

B. wait until suppression pool temperature reaches 95°F then control it per EP-3, Primary Containment Control.

C. suspend the RCIC surveillance if suppression pool temperature rises to >85°F.

D. wait until suppression pool temperature reaches 85°F then control it per EP-3, Primary Containment Control.

Answer: A

Explanation:

‘A’ is correct per 06-OP-1E51-Q-0003, P/L 2.3.

‘B’, ‘C’ and ‘D’ are incorrect for the reason that ‘A’ is correct. ‘B’ is plausible if all the candidate can recall is the EP-3 entry condition for supp pool temperature. ‘C’ and ‘D’ have face plausibility.

Technical References:

TS 3.6.2.1; 06-OP-1E51-Q-0003

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) and the containment spray (plant-specific): (CFR: 41.7 / 45.8).	Tier #	1
	Group #	1
	K/A #	295027 EK2.01
	Rating	3.2

Question 14

A LOCA has occurred.

The plant is operating in the "Unsafe to Initiate" part of the Containment Spray Initiation Pressure Limit (CSIPL) figure.

Containment sprays should <u>not</u> be initiated in this situation because there is the potential for:

A. thermal shock causing a failure of the containment spray header.

B. a rapid increase in containment pressure forcing water from the suppression pool into the drywell and a subsequent loss of ECCS injection.

C. losing all ECCS injection flow due to the loss of suppression pool inventory.

D. drawing a vacuum in containment and a loss of containment integrity.

Answer: D

Explanation:

A. This is incorrect because the water traveling the CS system from the suppression pool would be at an elevated temperature and there would be minimal thermal shock potential.

B. This is incorrect because while the containment is at a high temperature and low pressure, the sprays would not evaporate and increase pressure but would actually condense steam and reduce pressure.

C. This is incorrect because following a LOCA SP inventory would increase and any water used for CS would return to the suppression pool.

D. This is correct because per the EOP bases for Figure 3 of 05-S-01-EP-1, Containment Spray Initiation Pressure Limit (CSIPL), is the evaporative cooling would result in a rapid drop in pressure and could go sub-atmospheric and cause a loss of containment

integrity.

Technical References:

Procedure 05-S-01-EP-1 and the EP Technical Bases Document

References to be provided to applicants during exam:
None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret the Drywell pressure as it applies to HIGH DRYWELL TEMPERATURE (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295028 EA2.04
	Rating	4.1

Question 15

Use your provided references to answer this question.

Assuming the drywell is at saturation conditions and containment remains at 0 psig, the maximum drywell temperature that can be allowed without exceeding the drywell design pressure is approximately _____ °F.

A. 215

B. 260

C. 272

D. 282

Answer: C

Explanation:

A. This is incorrect because 215°F corresponds to 15.3 psia. Plausible to the Candidate who considers the maximum drywell pressure of 30 psig and subtracts (rather than adds) 14.7 psia to get 15.3 psia.

B. This is incorrect because i) of the discussion cited in the Explanation for Answer C, and ii) this is the maximum Drywell design differential pressure.

C. This is correct because the maximum drywell pressure is 30 psig or 44.7 psia. If the drywell is at saturation conditions, this equates to a saturation temperature of approximately 272°F.

D. This is incorrect because of the discussion cited in the Explanation for Answer C.

Technical References:

GLP-OPS-M4101, GLP-OPS-MCD07

References to be provided to applicants during exam:

Steam Tables

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret the Suppression pool temperature as it applies to LOW SUPPRESSION POOL WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295030 EA2.02
	Rating	3.9

Question 16

As suppression pool water level lowers due to a break in the Containment structure, suppression pool temperature elements will begin to uncover at:

A. 24.6 feet

B. 18.34 feet

C. 14.5 feet

D. 14.25 feet

Answer: D

Explanation:

A. See explanation for ‘D’.

B. See explanation for ‘D’.

C. See explanation for ‘D’.

D. This is correct per EP-1, CAUTION #2.

Technical References:

Procedures 05-S-01-EP-1, 05-S-01-EP-3

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to operate and/or monitor low pressure core spray as it applies to REACTOR LOW WATER LEVEL (CFR: 41.7 / 45.6).	Tier #	1
	Group #	1
	K/A #	295031 EA1.03
	Rating	4.4

Question 17

A small break LOCA has occurred in the drywell and currently:

- All ECCS are operating
- Drywell pressure is 1.50 psig and rising
- RPV level is -140 inches and lowering
- RPV pressure is 450 psig

Based on these conditions, the flow indicated on FI-R600, LPCS Pump Discharge Flow, would be expected to be approximately:

A. 0 gpm

B. 1250 gpm

C. 1380 gpm

D. 7200 gpm

Answer: A

Explanation:

A. This is correct because although LPCS would auto initiate, the discharge pressure would be less than RPV pressure resulting in no injection flow to the reactor vessel.

B. This is incorrect because there is no flow as discussed in A.

C. This is incorrect because there is no flow as discussed in A.

D. This is incorrect because there is no flow as discussed in A.

Technical References:

GLP-OPS-E2100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the reactor pressure effects on reactor power as it applies to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.8 to 41.10) (EK1.01)	Tier #	1
	Group #	1
	K/A #	295037 EK1.01
	Rating	4.1

Question 18

After operators insert a manual scram, the following conditions exist:

- Reactor power is 8% and stable
- Reactor pressure is stable at 900 psig on the bypass valves
- Suppression pool temperature is stable at 87°F
- RHR 'A' is operating in suppression pool cooling
- Reactor water level is -70" and slowly lowering
- Feedwater is available

MSIVs go closed due to loss of both RPS motor-generator sets.

CRS directs operators to lower reactor pressure to a new band of 450-600 psig using SRVs.

Which of the following describes the response of indicated reactor water level and reactor power as pressure is lowering?

A. Indicated level lowers; reactor power rises.

B. Indicated level rises; reactor power lowers.

C. Indicated level rises; reactor power rises.

D. Indicated level lowers; reactor power lowers.

Answer: B

Explanation:

As pressure lowers, voids begin to form, causing water to swell up into the downcomer, which results in a rise in indicated level. The same voiding adds negative reactivity, causing reactor power to lower.

For this reason, only 'B' is correct.

Technical References: EOP Bases Document

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for emergency depressurization as it applies to HIGH OFF-SITE RELEASE RATE (CFR: 41.5 / 45.6).	Tier #	1
	Group #	1
	K/A #	295038 EK3.04
	Rating	3.6

Question 19

Procedure 05-S-1-EP-4, Auxiliary Building Control, requires an emergency depressurization if a system is discharging outside of primary/secondary containment that cannot be isolated and a "General Emergency" declaration is expected due to a radioactive release.

The reason for emergency depressurizing is to:

A. Reduce the leak rate that is being released outside containment

B. Maintain adequate core cooling

C. Minimize the loss of RCS inventory

D. Ensure RPV level control is maintained using the preferred injection systems

Answer: A

Explanation:

A. This is correct because of the bases for entering Emergency Depressurization with a high radioactive release rate. The reason for taking this course of action is to place the primary system in the lowest possible energy state. This reduces the release rate potential to the outside.

B. This is incorrect because of the reason for entering the emergency depressurization stated for Answer A. That is the primary reason for entrance based on this situation.

C. This is incorrect because of the reason for entering the emergency depressurization stated for Answer A. That is the primary reason for entrance based on this situation.

D. This is incorrect because of the reason for entering the emergency depressurization stated for Answer A. That is the primary reason for entrance based on this situation.

Technical References:

EP Technical bases document 02-S-01-40, Attachment VII, page 15 of 16, step 11 discussion.

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for actions contained in the abnormal procedure for plant fire on site as it applies to PLANT FIRE ON SITE.	Tier #	1
	Group #	1
	K/A #	600000 – AK3.04
	Rating	2.8

Question 20

Due to troubleshooting in progress, all automatic CO_2 fire suppression actuations are disabled.

A fire starts in Room OC407, one of the RPS MG Set rooms.

As directed by procedure 10-S-03-2, Response To Fires, operators manually secure control building fan coil unit Z17-B002.

Per the 10-S-03-2 procedure, what is the reason for this action?

A. To aide in Fire Brigade accessibility to Room OC407.

B. To prevent smoke from migrating from Room OC407 to non-affected rooms.

C. To de-energize a significant piece of electrical equipment that could be damaged by the fire.

D. To keep CO_2 in Room OC407, should the Fire Brigade use portable CO_2 fire extinguishers.

Answer: B

Explanation:

A. This is incorrect; see explanation for ‘B’.

B. This is correct per 10-S-03-2, Section 6.2.2.f.

C. This is incorrect; see explanation for ‘B’.

D. This is incorrect; see explanation for ‘B’.

Technical References:

Procedure 10-S-03-2, “Response to Fires”

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	None	

Examination Outline Cross-Reference	Level	RO
Ability to identify and interpret diverse indications to validate the response of another indication (Loss of Main Condenser Vacuum) (CFR: 41.7 / 43.5 / 45.4).	Tier #	1
	Group #	2
	K/A #	295002 2.1.45
	Rating	4.3

Question 21

The reactor is at 80 percent power when the reactor operator observes main condenser vacuum indicating 26” Hg and degrading.

A degrading vacuum would be expected if:

A. Extraction steam to Feedwater Heater 5A has isolated.

B. Seal Steam Generator header pressure is 15 inches WC.

C. OG PRE-TREAT RAD MON DNSC alarm is actuated

D. SJAE A(B) STM SPLY FLO LO alarm is actuated

Answer: D

Explanation:

A. This is incorrect because although feedwater temperature would go down the temperature and pressure of the steam supply from the reactor would not change. Therefore this would not affect condenser vacuum (see GLP-OPS-N2335).

B. This is incorrect because although a decreasing seal steam generator header pressure that reaches 15” WC causes an alarm, it is not low enough the result in degrading vacuum.

C. This is incorrect because the pre-treat rad monitor has no isolation or actuation functions associated with it and would have no effect on vacuum.

D. This is correct because the reception of this alarm on Panel P680 will result in the automatic closing of isolation valve (F003) between the LP condenser and the in-service Steam Jet Air Ejectors, causing a loss of vacuum (GLP-OPS-N6200).

Technical References:

GLP-OPS-MCD12, GLP-OPS-N2335, GLP-OPS-N6200, GLP-OPS-N3300, GLP-OPS-D1721, GLP-OPS-N6465, ARI P601-19A-F7

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.43.5, 55.45.4	

Examination Outline Cross-Reference	Level	RO
Ability to determine and/or interpret Reactor water level as it applies to LOW REACTOR WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	295009 AA2.01
	Rating	4.2

Question 22

Current plant conditions are as follows:

- Reactor pressure is 1025 psig
- Recirc Pumps are in fast speed
- Actual reactor water level is +20”

Wide Range level should be indicating approximately ____”.

A. +35

B. +20

C. +5

D. -10

Answer: C

Explanation:

A. This is incorrect because based on the calibration conditions of the Wide Range level instrumentation (no recirculation flow), the level indication reads approximately 15” lower than actual level with the reactor recirc pumps running. Per GLP-OPS-B2101, Page 19, level reads about 15” lower than actual level. This would result in an indicated level of approximately +5”.

B. This is incorrect due to the effects described in the explanation to Answer A.

C. This is correct due to the effects described in the explanation to Answer A.

D. This is incorrect due to the effects described in the explanation to Answer A.

Technical References:

GLP-OPS-B2101, GLP-OPS-B3300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of temperature increases as it applies to HIGH DRYWELL PRESSURE (CFR: 41.8 to 41.10).	Tier #	1
	Group #	2
	K/A #	295010 AK1.03
	Rating	3.2

Question 23

The plant is at rated thermal power when Plant Service Water is lost to the drywell chillers.

In this situation, drywell pressure would be controlled by:

A. Placing the Containment Cooling System in the drywell purge mode of operation.

B. Manually initiating the Standby Gas Treatment System.

C. Manually initiating the Drywell Purge Subsystem of the Combustible Gas Control System.

D. Aligning SSW to the drywell chillers.

Answer: D

Explanation:

A. This is incorrect it is an acceptable means of manipulating Drywell pressure only in plant Modes 4 and 5.

B. This is incorrect because SGTS is not used for a backup function such as this. SGTS initiation isolations Containment Ventilation from interfacing with it, so its operation could not have any effect on Drywell conditions.

C. This is incorrect because this subsystem is used as a means to increase pressure in the Drywell and reduce Drywell hydrogen concentration during a LOCA. It does not have a backup function to help maintain normal Drywell pressure.

D. This is correct because the primary means of controlling drywell pressure is by controlling drywell temperature. This is done by using the Drywell Cooling and Drywell Chilled Water Systems. PSW is the primary cooling supply to the Drywell Chillers, with SSW as a backup.

Technical References:

GLP-OPS-M5100, GLP-OPS-M4100, GLP-OPS-T4801, GLP-OPS-E6100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8 to 55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the reasons for the increased containment cooling (Mark-III) response as it applies to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) (CFR: 41.5 / 45.6).	Tier #	1
	Group #	2
	K/A #	295011 AK3.01
	Rating	3.6

Question 24

Per EP-3, Containment Control, an emergency depressurization is required before reaching 185°F in the containment.

The reason for this is to:

A. keep the containment temperature below the design limit.

B. ensure the reactor pressure and water level instrumentation remain valid.

C. to ensure the RPV is depressurized while the suppression pool is still capable of absorbing the energy of a blowdown.

D. ensure the containment sprays can be used to cool the containment without violating the containment spray temperature limit.

Answer: A

Explanation:

A. This is correct because the primary containment design temperature limit of 185 degrees.

B. This is incorrect because although the instrumentation may be affected it will still be valid well above 185 degrees.

C. This is incorrect because the containment temperature is not related to blowdown capability. This is a function of suppression pool temperature.

D. This is incorrect because ability to use containment sprays is a function of both temperature and pressure. A lower temperature may still violate the CSIPL.

Technical References:

GLP-OPS-M4100, GLP-OPS-M4101 (Statement of Technical Specification 3.6.1.5),

GFIG-OPS-M4100, GLP-OPS-G3336

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the standby gas treatment/FRVS (CFR: 41.7 / 45.8).	Tier #	1
	Group #	1
	K/A #	295017 AK2.12
	Rating	3.4

Question 25

The Standby Gas Treatment System will automatically initiate if there is:

A. 20 mR/hr sensed at the Fuel Pool Sweep Exhaust

B. 3.5 mR/hr sensed at the Containment Vent Exhaust

C. 3.6 mR/hr sensed at the Fuel Handling Area Exhaust

D. 3 times the normal radiation level on any of the Main Steam Lines

Answer: C

Explanation:

A. This is incorrect because the threshold for SGTS trains starting based on FPS Exhaust radiation readings is 30 mR/hr.

B. This is incorrect because the radiation alarm set point for Containment Vent Exhaust does not provide an input to STGS.

C. This is correct this is one of the initiators for SGTS Trains “A” and “B” per GLP-OPS-T4801 and GLP-OPS-D1721.

D. This is incorrect because the radiation alarm set point for Main Steam Lines does not provide an input to STGS.

Technical References:

GLP-OPS-T4801, GLP-OPS-D1721, GFIG-OPS-D1721

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the Reactivity control concepts as they apply to LOSS OF CRD PUMPS: (CFR: 41.8 to 41.10).	Tier #	1
	Group #	2
	K/A #	295022 AK1.02
	Rating	3.6

Question 26

The running CRD pump trips and the crew enters the ONEP for Control Rod/Drive Malfunctions.

After starting the standby CRD pump and re-opening the CRD flow control valve, the operator mistakenly re-starts the tripped pump as well; the pump starts and remains running.

Per the Precautions/Limitations of the CRD System SOI, which of the following is a concern that is likely to result from this operator error?

A. Control rod drifts

B. Automatic trip of both CRD pumps

C. Failure of control rod drive piston seals

D. Water hammer damage to CRD system piping

Answer: A

Explanation:

A. This is correct as a concern identified in CRD SOI (04-1-01-C11-1), Precaution/Limitation 3.10. That P/L directs operators not to exceed a cooling water d/p of 20 psid; otherwise, rod drifts may occur.

B. This is incorrect; the dual CRD pump trip is neither a likely result of running two pumps, nor is it mentioned as a concern in the SOI P/Ls.

C. This is incorrect; CRD piston seal failure is neither a likely result of running two pumps, nor is it mentioned as a concern in the SOI P/Ls.

D. This is incorrect; CRD piston seal failure is neither a likely result of running two pumps, nor is it mentioned as a concern in the SOI P/Ls.

Technical References:

GLP-OPS-C111A, GFIG-OPS-C111A, ONEP 05-1-02-IV-1, CRD SOI 04-1-01-C11-1

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8, 55.41.9, 55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine and / or interpret the Hydrogen monitoring system availability as it applies to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	500000 EA2.01
	Rating	3.1

Question 27

Plant conditions are as follows:

- Drywell pressure is 4.5 psig
- Bus 16AB is locked out
- Division I ECCS has failed to initiate manually and automatically
- Reactor water level is stable at -22" Wide Range

Per EP-3, Containment Control, what is the required operator action for Hydrogen Control?

A. Start the Division II CTMT and Drywell Hydrogen Analyzers by placing their handswitches in START.

B. Sample the Containment for hydrogen in accordance with 08-S-04-957.

C. Exit all EPs and enter all SAPs.

D. Start the Division I CTMT and Drywell Hydrogen Analyzers by placing their handswitches in START.

Answer: D

Explanation:

A. This is incorrect because the Div II analyzers cannot be energized with the bus 16AB lockout.

B. This is incorrect because sampling is required only if the analyzers are unavailable. They can be manually started.

C. This is incorrect because; no SAP entry is required until hydrogen is >2.9%; also, reactor water level is stable well above TAF.

D. This is correct because EP-3 requires the crew to verify the analyzers are operating. In this case, they will have to manually start them.

Technical References:

Procedure 05-S-01-EP-3, Primary Containment Control

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc. (CFR: 41.10 / 45.1 / 45.12) (2.1.29)	Tier #	2
	Group #	1
	K/A #	Generic – 2.1.29
	Rating	4.1

Question 28

A valve associated with a tagout is located in a High Radiation Area (HRA).

Per EN-OP-102, Protective and Caution Tagging, which of the following describes the requirement/allowance for hanging a DANGER tag on the valve handwheel and for verifying the valve is in the required position?

A. Must enter the HRA area under an approved RWP to hang the tag and verify valve position, without exception.

B. May hang the tag for the valve at the entrance to the HRA; no valve position verification is required.

C. Neither hanging the tag nor verifying valve position is required if the valve will not have work performed on it and it will remain inaccessible for the duration of the tagout.

D. Hanging a tag is not required if an alternate method can be used to verify valve position, the valve will not have work performed on it, and it will remain inaccessible during the tagout implementation.

Answer: D

Explanation:

A. This is incorrect because it is not mandatory to enter area of the valve to apply the tag. There are exception criteria stated in Section 2.5 of Attachment 9.2, EN-OP-102.

B. This is incorrect because the method stated involving applying a tag to the entrance to the area containing the valve involves a Caution tag vice a Danger tag.

C. This is incorrect because it only meets a portion of one of the methods of addressing the tagout in EN-OP-102, Attachment 9.2. Verification of valve position must be provided for if this approach is used, and it is not stated as part of the answer.

D. This is correct because it meets acceptance criteria for an exception to hanging a

tag/conducting valve lineup verification per Section 2.5 of Attachment 9.2, EN-OP-102. Having the valve in a High Radiation Area qualifies as an inaccessible area.

Technical References:

EN-OP-102, Attachment 9.2

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on reactor water level (plant-specific): (CFR: 41.7 / 45.4) (K3.02 – RO)	Tier #	2
	Group #	1
	K/A #	205000 K.2
	Rating	3.2

Question 29

The plant is in Mode 4 with the following:

- RHR A is operating in Shutdown Cooling using the feedwater header return line
- Reactor Recirc Pump A is operating
- Feed and Condensate are not in service

A scaffold pole falls on the MOV for E12-F053A, RHR A Shutdown Cooling Return to Feedwater, and E12-F053A closes.

Without operator action, which of the following will occur first when E12-F053A closes?

A. Reactor water level will lower.

B. RHR Pump A will trip.

C. CST level will rise.

D. Reactor water level will rise.

Answer: A

Explanation:

A. When the shutdown cooling return valve closes the min flow valve (E12-F064A) will open due to the low flow condition. This will result in water being pumped from the RPV to the suppression pool. Consequently, RPV level will lower and suppression pool level will rise.

B. This is incorrect because it is not the first response. The RHR pump remains running until RPV level lowers to the Group 3 auto-isolation setpoint of +11.4”, at which time the SDC common suction valves (F008 and F009) auto-close, then causing an RHR Pump trip.

C. is incorrect; however, the candidate may choose this answer if the student believes the min-flow valve discharges to the CST.

D. is incorrect because water inventory is being lost through the min-flow valve. Some candidates may choose this answer if they don't consider the opening of the min flow valve or the new discharge flow-path. If these are not considered, indicated level would slowly go up due to decay heat induced swell.

Technical References:

Lesson Plan GLP-OPS-E1200

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-E1200, objectives 3.4, 8.1, and 8.2

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the process for conducting special or infrequent tests as it relates to LPCS. (CFR: 41.10 / 43.3 / 45.13) (2.2.7)	Tier #	2
	Group #	1
	K/A #	209001 G2.2.7
	Rating	2.9

Question 30

A modification to the Low Pressure Core Spray motor windings requires a special test be performed by electrical maintenance before it is declared operable.

Prior to performing the Special Test Instruction (STI), procedure 01-S-07-2, Test Control, requires a Pre-Test Briefing be conducted by the:

A. Individual supervising the test.

B. Control Room Supervisor.

C. Shift Manager.

D. Superintendent of Electrical Maintenance.

Answer: A

Explanation:

A. This is correct because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.

B. This is incorrect because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.

C. This is incorrect because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.

D. This is incorrect because the referenced procedure requires the Pre-Test Briefing by held by the person supervising the test.

Technical References: Procedure 01-S-07-2, Test Control

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the Suppression pool suction strainer (BWR-5, 6) will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	209002 K6.04
	Rating	2.5

Question 31

During the last refueling outage, the old “cone” suction strainer (in the suppression pool) for HPCS was blank-flanged.

HPCS is in service taking a suction from the suppression pool ring suction strainer and discharging back to the suppression pool for a post maintenance test.

During the test, the HPCS suction becomes blocked by a large sheet of plastic that has enveloped that section of the suppression pool ring suction strainer.

One of the effects of this blockage would be HPCS:

A. pump suction pressure will rise.

B. motor current will lower.

C. pump discharge pressure will rise.

D. motor current will rise.

Answer: B

Explanation:

A. This is incorrect because HPCS pump suction pressure will decrease.

B. This is correct because the amount of work being performed by the pump decreases as flow goes down. The reduction in pump work will cause a decrease in motor current.

C. This is incorrect because as pump flow and suction pressure go down, pump discharge pressure will also go down.

D. See explanation for answer B.

Technical References:

GLP-OPS-E2201

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on the core spray line break detection system (plant-specific) (CFR: 41.7 / 45.4).	Tier #	2
	Group #	1
	K/A #	211000 K3.02
	Rating	3.0

Question 32

The plant is operating at rated power when the SLC Manual Isolation Valve, C41-F008 (in the drywell), develops a significant amount of valve body leakage.

This condition could result in a:

A. RCIC leakage alarm.

B. HPCS out of service alarm.

C. RPV head leakage alarm.

D. RWCU line break alarm.

Answer: B

Explanation:

A. This is incorrect because the SLC system does not interface with the RCIC system.

B. This is correct because the SLC discharge line is connected to the HPCS discharge line downstream of the HPCS testable check valve. Reducing the pressure sensed in this line compared to the pressure sensor located above the Core Plate will result in an increased differential pressure. A differential pressure of +/- 1.2 PSID between these points will result in both a LINE BREAK status light on the HPCS status panel, and a HPCS SYS OOSVC alarm will annunciate.

C. This is incorrect because the SLC system does not interface with the RPV head.

D. This is incorrect because the SLC system does not interface with the RWCU system.

Technical References:

GLP-OPS-E3100, GFIG-OPS-E3100, GFIG-OPS-E2201, GLP-OPS-C4100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.4	

Examination Outline Cross-Reference	Level	RO
Knowledge of electrical power supplies to the RPS motor-generator sets (CFR: 41.7).	Tier #	2
	Group #	1
	K/A #	212000 K2.01
	Rating	3.2

Question 33

The electrical power supply to RPS Motor Generator Set "B" is:

A. 13B22

B. 14B22

C. 15B42

D. 16B22

Answer: B

Explanation:

A. This is incorrect because it is not the correct power supply.

B. This is correct because it is the power supply to the RPS "B" motor generator set.

C. See Explanation for Answer B.

D. See Explanation for Answer B.

Technical References:

GLP-OPS-C7100 and 04-1-01-C71-1, Attachment III, page 1 of 3.

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the rod withdrawal blocks (CFR: 41.7).	Tier #	2
	Group #	1
	K/A #	215003 K4.01
	Rating	3.7

Question 34

The reactor is in Mode 2 with a startup in progress.

All of the Intermediate Range Monitors (IRMs) are indicating mid-scale on Range 5 when IRM "A" fails downscale.

Which of the following describes the impact of this IRM failure?

A. A rod withdraw block ONLY is in effect.

B. A scram will occur if the IRM B switch is taken out of "OPERATE"

C. A rod withdraw AND insert block are in effect

D. Control rods can still be inserted or withdrawn

Answer: A

Explanation:

A. This is correct because with any one IRM downscale on range 5 a rod withdraw block is in effect.

B. This is incorrect because the IRM downscale does not create a scram signal. Therefore taking IRM B out of operate will result in a half scram.

C. This is incorrect because the IRM rod blocks do not affect the ability to insert rods.

D. This is incorrect because a single downscale signal causes a withdraw rod block.

Technical References:

GLP-OPS-C5102, GFIG-OPS-C5102

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the RPS will have on the SOURCE RANGE MONITOR (SRM) SYSTEM (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	215004 K6.01
	Rating	3.2

Question 35

The plant is in Mode 5.

The RPS shorting links are removed.

The Electrical Protection Assembly (EPA) associated with RPS Bus 'A' motor-generator trips on undervoltage.

Which of the following describes the immediate response of SRM 'A' to the EPA trip?

SRM 'A' will:

A. De-energize; the de-energized SRM will generate a full scram.

B. De-energize; the de-energized SRM will generate a half-scram.

C. Remain energized; the SRM will generate a half-scram.

D. Remain energized; the SRM will not generate a scram signal (full or half).

Answer: D

Explanation:

A. This is incorrect; see explanation for 'D'.

B. This is incorrect; see explanation for 'D'.

C. This is incorrect; see explanation for 'D'.

D. This is correct because the SRMs are powered from inverters, not from RPS. Loss of RPS Bus 'A' power (when the EPA trips) does not affect SRM 'A'. SRM 'A' remains fully energized and has no reason to generate a trip output to RPS logic; so the shorting links being removed is irrelevant here. Although the EPA trip does amount to a half-scram (RPS 'A), this has nothing to do with the status of SRM 'A'.

Technical References:

GLP-OPS-C7100, GLP-OPS-C5101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of assignment of LPRM's to specific APRM channels as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM (CFR: 41.5 / 45.3).	Tier #	2
	Group #	1
	K/A #	215005 K5.06
	Rating	2.5

Question 36

LPRM detectors are assigned to a specific APRM channel in order to:

A. maintain axial and rotational power symmetry.

B. maintain rotational and mirror-image power symmetry

C. minimize wiring function problems associated with the detectors observed based on industry experience

D. ensure that each APRM channel receives an adequate representation of the local flux throughout the core, both radially and axially.

Answer: D

Explanation:

A. This is incorrect because it ties into the basis for LPRM placement in the core (symmetrical monitoring method), but it is not the basis for the assignment of the LPRM detectors to the APRMs. The axial element addresses part of the LPRM/APRM association issue, but it is out of context with the rest of the answer.

B. This is incorrect it ties into the basis for LPRM placement in the core (symmetrical monitoring method), but it is not the basis for the assignment of the LPRM detectors to the APRMs.

C. This is incorrect because there is no industry feedback that would serve as the major reason for not changing LPRM detector assignments.

D. This is correct because it is the basis stated in GLP-OPS-C5103, Page 15, and GLP-OPS-C5104, Page 10.

Technical References:

GLP-OPS-C5104, GLP-OPS-C5103

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.45.3	

Examination Outline Cross-Reference	Level	RO
Knowledge of electrical power supplies to the gland seal compressor (vacuum pump): (CFR: 41.7)	Tier #	2
	Group #	1
	K/A #	217000 K2.04
	Rating	2.6

Question 37

The power supply to the RCIC gland seal compressor is DC Power Panel:

A. 1DA2

B. 1DB2

C. 1DC2

D. 1DD2

Answer: A

Explanation:

A. This is correct because it is the power supply to the RCIC gland seal compressor per GLP-OPS-E5100.

B. This is incorrect because it is not the correct power supply.

C. See Explanation for Answer B.

D. See Explanation for Answer B.

Technical References:

GLP-OPS-E5100, GLP-OPS-E2201, 04-1-01-E51 Attachment III, page 5.

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the condensate storage and transfer system will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	217000 K6.04
	Rating	3.5

Question 38

The normally locked-open valve, P11-F021 (CST Supply To HPCS/RCIC) is currently seized on its closed seat and will not open.

What impact does this have on the RCIC system?

A. No water is available for flushing the system.

B. Full-flow tests cannot be performed on the system.

C. The pump has no minimum flow line discharge path.

D. The pump has only its normal suction source available.

Answer: B

Explanation:

A. This is incorrect because the CST does not provide a flushing water volume for the RCIC system. The CST provides flushing capability for other systems (see GFIG-OPS-P1100, Figure 1), but not for the RCIC system.

B. This is correct because RCIC uses only the CST for flow testing.

C. This is incorrect because the RCIC pump min-flow discharges to the Supp Pool, not to the CST.

D. This is incorrect because the RCIC pump's normal suction source is the CST; its alternate (backup) suction is from the Supp Pool.

Technical References:

GLP-OPS-E5100, GFIG-OPS-E5100, GLP-OPS-P1100, GFIG-OPS-P1100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the nuclear boiler instrument system (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	1
	K/A #	218000 K1.03
	Rating	3.7

Question 39

A major reactor coolant system leak begins in the drywell.

Ten minutes later, plant conditions are as follows:

- Reactor water level is -140” and lowering slowly
- Drywell pressure is 1.3 psig
- RPV pressure is 650 psig
- RCIC is injecting into the RPV

A control room operator manually starts the LPCS pump.

Assuming ADS has <u>not</u> been inhibited, the ADS:

A. 105-second timer is still reset (i.e., has not yet started).

B. 105-second timer started immediately following the LPCS pump start.

C. valves opened immediately following the LPCS pump start.

D. valves are still closed but will immediately open when RPV level reaches Level 1.

Answer: A

Explanation:

A. This is correct because the Level 1 initiation signal has not been satisfied.

B. This is incorrect the 105 second time delay does not start until three indications are provided (reactor vessel ≤ +11.4”, and reactor vessel ≤ -150.3”, and Drywell pressure ≥ 1.39 psig, or at or below -150.3” for 9.2 minutes).

C. This is incorrect because although one low pressure ECCS pump must be running, ADS will not initiate until RPV level reaches Level 1 with 9.2 minute time delay, or >1.39 psig in DW, plus 105 sec time delay.

D. This is incorrect because when level 1 is reached, the 9.2 minute timer second time

delay will initiate; otherwise, must have >1.39 psig in DW to start the 105-second timer.

Technical References:

GLP-OPS-E2202

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the Loss of A.C. or D.C. power to ADS valves on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6).	Tier #	2
	Group #	1
	K/A #	218000 A2.05
	Rating	3.4

Question 40

A simultaneous loss of LCC 15BA3 and LCC 15BA6 occurs.

In each case, the loss is due to an internal fault on that LCC.

Which of the following describes how this event impacts ADS, and identifies Technical Specification entry requirements resulting from this event?

A. Div I and Div II ADS logics will <u>not</u> initiate if required.
ADS valve 'A' and 'B' solenoids still have a source of power.
Entry into LCO 3.5.1, ECCS – Operating, is required.

B. Div I ADS logic will initiate if required.
ADS valve 'A' solenoids still have a source of power.
Entry into LCO 3.8.4, DC Sources – Operating, is required.

C. Div I ADS logic will <u>not</u> initiate if required.
ADS valve 'A' solenoids do <u>not</u> have a source of power.
Entry into LCO 3.5.1, ECCS – Operating, <u>and</u> into LCO 3.8.4, DC Source – Operating, is required.

D. Div II ADS logic will <u>not</u> initiate if required.
ADS valve 'B' solenoids do <u>not</u> have a source of power.
Entry into LCO 3.5.1, ECCS – Operating, <u>and</u> into LCO 3.8.4, DC Source – Operating, is required.

Answer: B

Explanation:

Loss of both LCCs takes away both chargers for Div I battery bus 11DA. However, because the battery bank itself is still connected to the 11DA bus, all of the Div I ADS logic still has its required DC logic initiation power. Similarly, the battery itself is able to power the ADS valve 'A' solenoids if required.

Since the ADS system is unaffected by this event, there is no LCO 3.5.1 entry required. However, the loss of both 11DA bus chargers does require an entry into LCO 3.8.4 for

implementation of ACTION A.1.

For all of these reasons, ‘B’ is correct and ‘C’ is incorrect.

‘A’ and ‘D’ are incorrect because these two LCCs impact only Div I, not Div II.

Technical References:

GLP-OPS-E2202, GLP-OPS-L1100, TS 3.5.1 and 3.8.4
References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including valve closures: (CFR: 41.5 / 45.5).	Tier #	2
	Group #	1
	K/A #	223002 A1.02
	Rating	3.7

Question 41

RPV level is +20 inches and slowly lowering.

As RPV level continues to slowly lower, the first automatic isolation would be the:

A. Main steam isolation valves (MSIVs)

B. Reactor Water Cleanup isolation valves

C. RHR to Radwaste isolation valves

D. Service air isolation valves

Answer: C

Explanation:

A. This is incorrect because the NSSSS isolation would occur at level 1 (- 150 incles).

B. This is incorrect because the NSSSS isolation would occur at level 2 (- 41.6 inches).

C. This is correct because the isolation would occur at + 11.4 inches. Of the 4 choices, this would be the first setpoint reached as level decreased.

D. This is incorrect because this isolation would occur at level 2 (- 41.6 inches).

Technical References:

GLP-OPS-M7101, 05-1-02-III-5, ONEP

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.45.5	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS Actuation (CFR: 41.5 / 45.6) (A2.04)	**Tier #**	1
	Group #	1
	K/A #	239002 A2.04
	Rating	4.1

Question 42

The plant is operating at rated power with no work activities in progress.

Six (6) ADS valves unexpectedly open and will not re-close.

Current plant conditions are:

- HI-HI Levels in Feedwater Heaters 6A and 6B
- Feedwater inlet temperature is lowering
- There is a Level 8 scram signal on RPS ‘B’ due to the initial RPV water level swell when the ADS valves opened
- Total core flow has risen 4%
- APRM Upscale alarms have annunciated
- Reactor power is stable
- Suppression pool temperature is 109°F and rising rapidly

Based on these conditions, the **NEXT** operator action should be:

A. Take manual control of feedwater and return RPV water level to the normal band.

B. Reduce Reactor Recirc flow to reduce reactor power to less than 100%.

C. Place the Mode Switch in SHUTDOWN to preclude exceeding the Technical Specification limit of 110°F.

D. Take the immediate actions of the Loss of Feedwater Heating ONEP.

Answer: C

Explanation:

A. This is incorrect because there is no indication of a problem with the feedwater level control system.
B. This is incorrect because with suppression pool temperature rapidly rising with no success path to maintain temperature below the TS limit, reducing reactor power would serve no purpose.
C. This is correct because with suppression pool temperature at 109 degrees, rapidly

rising, and no means to close the ADS valves, the TS limit of 110 degrees will be exceeded. The TS required action is to place the Mode Switch in shutdown.

D. This is incorrect because the suppression pool temperature will be exceeded before feedwater temperature can be restored and this requires placing the Mode Switch in Shutdown.

Technical References: OE 772 – Manual Scram Due to Multiple Lifts of SRV Valves at Grand Gulf

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including runout flow control (plant-specific): (CFR: 41.7 / 45.7) (A3.01)	Tier #	1
	Group #	1
	K/A #	259002 A3.01
	Rating	3.0

Question 43

The following plant conditions exist:

- Reactor Feed Pump 'A' flow is 5 mlbm/hr
- Reactor Feed Pump 'B' is being placed in service and its flow is 3.5 mlbm/hr
- DFCS is in 3-element control
- Reactor water level is +40"
- Both Reactor Recirc Pumps are in fast speed

Following a trip of Reactor Feed Pump 'A', reactor water level drops to +35" before recovering.

This feedwater transient will cause:

A. Reactor recirc pumps to downshift to slow speed.

B. Reactor recirc flow control valves to ramp to the minimum flow position.

C. DFCS to shift to single-element control.

D. DFCS flow signal to transfer from normal to estimated.

Answer: C

Explanation:

A. This is incorrect because feedwater flow must drop below 3.0 mlbm/hr for 15 seconds to cause a downshift of the Recirc pumps (i.e., the "Recirc Pump Cavitation Downshift Interlock"). RFP 'B' flow is still at least 3.5 mlbm/hr.

B. This is incorrect because is suggests the Recirc flow control valve runback. Even with the feed pump trip, this won't occur because level never drops to 32.7" in the stem.

C. This is correct because DFCS 3-element is de-selected (leaving single-element in control) whenever total feed flow drops below 4.95 mlbm/hr. With the feed pump trip, RFP 'B' flow is only 3.5 mlbm/hr.

D. This is incorrect because DFCS control transfers to an ”estimated” flow signal only if one of the flow signal inputs fails.

Technical References: Lesson Plans C34 and B33

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including system flow (CFR: 41.5 / 45.5).	Tier #	2
	Group #	1
	K/A #	261000 A1.01
	Rating	2.9

Question 44

The following plant conditions exist:

- Reactor water level is -44”
- Drywell pressure is 1.20 psig
- Standby Gas Treatment System Trains “A” and “B” are in AUTO

The hand switch for SGTS Train “B” exhaust fan is then placed in the STOP position and no other SGTS-related controls are manipulated.

The resulting steady-state SGTS system total exhaust flow (i.e., considering both trains) is approximately:

A. 0 scfm

B. 1850 scfm

C. 3850 scfm

D. 7700 scfm

Answer: D

Explanation:

A. This is incorrect because the Train “B” exhaust fan would not secure from operation. See Explanation for Answer D.

B. This is incorrect based on the Explanation for Answer D.

C. This is incorrect and is based on only one fan running. See the Explanation for Answer D.

D. This is correct because with the SGTS initiation signal present (-41.6”), placing the exhaust fan control hand switch for either train in the STOP position will not secure that fan. Therefore, both fans will continue to operate and maintain the discharge

flow that it is set to provide in normal operation. The approximate system flow-rate is 3875 scfm per fan; i.e., ~7700 scfm total system flow.

Technical References:

GLP-OPS-T4801

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: principle involved with paralleling two A.C. sources (CFR: 41.5 / 45.3).	Tier #	2
	Group #	1
	K/A #	262001 K5.01
	Rating	3.1

Question 45

An operator is paralleling the Diesel Generator 12 (DG 12) with Bus 16AB for a post maintenance test.

The operator is about to close the output breaker for DG 12 with the synchroscope rotating slowly in the counter-clockwise direction.

Closing the breaker in this situation could result in DG 12:

A. Tripping on over-current.

B. Tripping on reverse power.

C. Picking up excessive positive MVARs.

D. Picking up excessive negative MVARs.

Answer: B

Explanation:

A. This is incorrect because the reverse power trip would cause the DG to trip first with a negative power being assumed by the DG.

B. This is correct because with the synch scope rotating in the slow direction (counter clockwise) with the DG as the incoming generator, the bus would pick up the DG as a load. This would result in a reverse power condition and the DG would trip on the reverse power trip.

C. This is incorrect because the synchroscope is used to match real load and does not affect the reactive loading of the generator.

D. This is incorrect because the synchroscope is used to match real load and does not affect the reactive loading of the generator.

Technical References:

GLP-OPS-R2100, GFIG-OPS-R2100, GLP-OPS-P7500

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.3	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor in the control room: voltage, current, power, and frequency on A.C. buses (CFR: 41.7 / 45.5 to 45.8)	Tier #	2
	Group #	1
	K/A #	262001 A4.05
	Rating	3.3

Question 46

The Division 3 Diesel Generator (DG) is carrying the Division 3 electrical distribution system following a loss of off-site power.

Placing the DG speed control switch in the RAISE position will cause generator _____ to rise.

A. MW

B. MVARS

C. voltage

D. frequency

Answer: D

Explanation:

A. This answer is incorrect because with the diesel operating alone the MW loading is strictly a function of the loads operating on the bus and raising speed will have no affect on the operating loads.

B. This answer is incorrect because with the diesel operating alone the MVAR loading is strictly a function of the types of loads operating on the bus and raising speed will have no affect on the operating loads.

C. This answer is incorrect because operating the engine speed control will have no affect on generator voltage.

D. This answer is correct because with the diesel operating alone the engine speed control will change the speed of the engine and thereby change the operating frequency of the generator.

Technical References:

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: transfer from preferred power to alternate power supplies (CFR: 41.7) (K4.01)	Tier #	2
	Group #	1
	K/A #	262002 K4.01
	Rating	3.1

Question 47

An equipment operator has been directed to manually transfer a BOP static inverter from the normal to the alternate source of power.

When the manual static transfer switch is depressed, the inverter will transfer to the alternate power source only if the:

A. INVERTER FAILURE indicator is lit.

B. REVERSE POLARITY indicator is not lit.

C. IN SYNC indicator is lit.

D. MANUAL BYPASS SWITCH is in the BYPASS position.

Answer: C

Explanation:

A. This is incorrect because this indicator indicates a transfer should have occurred but does not have any input to a manual transfer.
B. This is incorrect because this indicator indicates the power inputs to the inverter are reversed but this will not prevent a transfer.
C. This is correct because if the two power sources are not in synch the transfer will not occur.
D. This is incorrect because taking this switch to bypass will immediately transfer the power source regardless of the static transfer switch.

Technical References: Lesson Plan L62, page 14

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor [UPS] in the control room: transfer from alternative source to preferred source (CFR: 41.7 / 45.5 to 45.8) (A4.01)	Tier #	2
	Group #	1
	K/A #	262002 A4.01
	Rating	2.8

Question 48

BOP inverter 1Y80 has automatically transferred from its alternate source to its normal source.

What control room alarm will alert the operators of this transfer?

A. SPDS CMPTR TROUBLE alarm annunciates.

B. STATIC INVRTR 1Y80 TROUBLE alarm annunciates.

C. SPDS CMPTR TROUBLE alarm clears.

D. STATIC INVRTR 1Y80 TROUBLE alarm clears.

Answer: D

Explanation:

There is no SPDS CMPTR TROUBLE alarm; thus, A and C are incorrect.

With the inverter having been already on its alternate, the STATIC INVRTR 1Y80 TROUBLE alarm is already sealed-in. It clears when its transfers back to normal. For this reason, ‘B’ is incorrect and ‘D’ is correct.

Technical References: ARI P807-3A-H1; 04-1-01-L62-1, section 4.1

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and Battery charger and battery: (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	1
	K/A #	263000 K1.02
	Rating	3.2

Question 49

A loss of ESF power panel 15P42 would result in a loss of charging capability for battery bank:

A. 1A3

B. 1D3

C. 1H1

D. 1K3

Answer: C

Explanation:

A. This is incorrect because battery bank 1A3 is associated with ESF DC battery chargers 1A4 and 1A5 which are 480 VAC chargers.

B. This is incorrect because battery bank 1D3 is associated with ESF DC battery chargers 1D4 and 1D5 which are 480 VAC chargers.

C. This is correct because battery bank 1H1 (+24 VDC) is associated with battery charger 1H4 which is a 120 VAC charger. ESF power panel 15P42 (120 VAC) powers this charger.

D. This is incorrect because battery bank 1K3 is associated with BOP DC battery chargers 1K4 and 1K5 which are 480 VAC chargers.

Technical References:

GLP-OPS-L1100, GFIG-OPS-L1100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for speed droop control (CFR: 41.7).	Tier #	2
	Group #	1
	K/A #	264000 K4.03
	Rating	2.5

Question 50

Diesel Generator 11 (DG 11) is running in parallel with the grid during its monthly surveillance.

During the surveillance, the operator inadvertently places the Unit Parallel Switch in RESET.

In response to this, the DG 11 output breaker will:

A. remain closed and the voltage regulator will shift to the isochronous mode.

B. immediately trip open.

C. immediately trip open ONLY if there is a LOCA signal present.

D. remain closed and the voltage regulator will remain in the droop mode.

Answer: B

Explanation:

A. This is incorrect because placing the switch in the reset position will immediately trip the generator output breaker.

B. This is correct because placing the switch in the reset position will immediately trip the generator output breaker.

C. This is incorrect because placing the switch in the reset position will immediately trip the generator output breaker.

D. This is incorrect because placing the switch in the reset position will immediately trip the generator output breaker.

Technical References:
GLP-OPS-P7500.16, Page 45 of 73

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the connections and / or cause- effect relationships between INSTRUMENT AIR SYSTEM and the cooling water to compressor (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	1
	K/A #	300000 K1.04
	Rating	2.8

Question 51

On a loss of off-site power, the alignment of cooling water for the Unit 1 Instrument Air Compressors shifts as follows:

A. TBCW cooling is lost, and SSW Loop B is automatically aligned to cool the compressor.

B. CCW cooling is lost, and PSW is automatically aligned to cool the compressor.

C. TBCW cooling is lost, and PSW is automatically aligned to cool the compressor.

D. CCW cooling is lost, and SSW Loop A is automatically aligned to cool the compressor.

Answer: A

Explanation:

A. This is correct because this is the automatic realignment for Loss of Power in GLP-OPS-P5300.

B. This is incorrect because neither CCW nor PSW has any direct relationship with cooling the air compressors.

C. This is incorrect because PSW has no direct relationship with cooling the air compressors.

D. This is incorrect because neither CCW nor SSW Loop A has any direct relationship with cooling the air compressors.

Technical References:

GLP-OPS-P5300, GFIG-OPS-P5300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including air temperature (CFR: 41.7 / 45.7).	Tier #	2
	Group #	1
	K/A #	300000 A3.02
	Rating	2.9

Question 52

The Unit 1 Instrument Air Compressor is running when the following alarm lights illuminate at the compressor's local control panel:

- High Discharge Air Temperature
- High Condensate Level

Which of the following describes the status of the Unit 1 Instrument Air Compressor?

A. It has tripped due to high discharge air temperature.

B. It has tripped due to high intercooler/aftercooler drain trap levels.

C. It is running with an elevated discharge air temperature.

D. It is running with elevated intercooler/aftercooler shell-side levels.

Answer: A

Explanation:

A. This is correct because the high air discharge temperature alarm is associated with an automatic compressor trip function per ARI P870-7A-A3.

B. This is incorrect because although there is an alarm condition on the condensate levels in the compressor, there is no automatic compressor trip function associated with this alarm.

C. See the Explanation for Answer A.

D. See the Explanation for Answer A.

Technical References:

ARI P870-7A-A3

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of tagging and clearance procedures (CFR: 41.10 / 45.13).	Tier #	2
	Group #	1
	K/A #	400000 General – 2.2.13
	Rating	4.1

Question 53

For which of the following activities must a DANGER tagout be used per EN-OP-102, Protective and Caution Tagging?

A. Working on an energized electrical circuit with voltage reading 24 VDC to ground

B. Isolating a CRD accumulator to recharge it using the CRD System SOI

C. Using a mechanical work package to internally inspect a check valve for preventive maintenance

D. Working on a temporary sump pump that is powered from an electrical outlet via a plug

Answer: C

Explanation:

A. This is incorrect per the exemption described in EN-OP-102, section 5.2[4].
B. This is incorrect per the exemption described in EN-OP-102, section 5.2[3], 3rd bullet.
C. This is correct; there is no EN-OP-102 exemption for this activity.
D. This is incorrect per the exemption described in EN-OP-102, section 5.2[3], 1st bullet.

Technical References:

EN-OP-102, Attachment 9.2

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.45.13	

Examination Outline Cross-Reference	Level	RO
Ability to manually operate and/or monitor the SDV isolation valve test switch in the control room (CFR: 41.7 / 45.5 to 45.8).	Tier #	2
	Group #	2
	K/A #	201001 A4.06
	Rating	2.8

Question 54

The ACRO is testing the operation of the CRD SDV Vent & Drain valves.

The ACRO simultaneously depresses and holds both pushbuttons (C11-F009A, CRD DISCH VOL VENT/DR SOV TEST and C11-F009B, CRD DISCH VOL VENT/DR SOV TEST).

The ACRO should expect to observe the following lights illuminated:

	CRD DISCH VOL VENT	CRD DISCH VOL DR
A.	green	green
B.	red	green
C.	red	red
D.	green	red

Answer: A

Explanation:

Holding both pushbuttons depressed de-energizes both solenoids (RPS ‘A’ solenoid and RPS ‘B’ solenoid) for SDV Vent & Drain Pilot Valve F009, causing F009 to vent air off of its associated SDV Vent & Drain valves (F010 and F011, respectively), which closes the valves. With F010 and F011 closed, their red lights are extinguished and their green lights are illuminated.

For this reason, only ‘A’ is correct.

Technical References:

GLP-OPS-C111A, GFIG-OPS-C111A

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.5, 55.45.6, 55.45.7, 55.45.8	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROD AND DRIVE MECHANISM and the control rod drive hydraulic system: (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	2
	K/A #	201003 K1.01
	Rating	3.2

Question 55

Control rod withdrawal speed would increase if the control rod drive hydraulic:

A. Flow control valve fails open

B. Pressure control valve fails open

C. Associated stabilizer valve fails closed

D. Cooling/exhaust water pressure equalizing valve fails closed

Answer: A

Explanation:

A. This is correct because it would provide for increased drive water pressure to the CRD mechanisms, causing control rod speed to increase.

B. This is incorrect because CRD drive pressure would decrease, causing control rod speed to decrease.

C. This is incorrect because the associated stabilizer valve does go closed on a withdrawal. After withdraw, a failed-closed stab valve becomes meaningless.

D. This is incorrect because these valves are normally closed at power operations.

Technical References:

GLP-OPS-C111B, GFIG-OPS-C111A, GLP-OPS-C111A, GFIG-OPS-B1300

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2 through 55.41.9; 55.45.7 and 55.45.8	

Examination Outline Cross-Reference	Level	RO
RCIS - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)	**Tier #**	3
	Group #	
	K/A #	201005 - 2.2.44
	Rating	4.2

Question 56

Following a malfunction with the rod position indicator for control rod 28-33, the rod is bypassed in the Rod Action Control System (RACS).

Verification that the correct control rod has been bypassed could be performed by:

A. ensuring rod 28-33 will not move when selected for individual rod movement and given a withdraw command.

B. ensuring the POS BYP pushbutton on P680 is lit.

C. depressing the POS BYP pushbutton on P680 and ensuring rod 28-33 has a green indicator lit on the full core display.

D. depressing the SUBST POS pushbutton on P680 and ensuring rod 28-33 has a red indicator lit on the full core display.

Answer: C

Explanation:

A. This is incorrect for bypassing the rod in the RACS but would be correct if the rod is bypassed in the RGDS.

B. This incorrect because any control rod being bypassed in the RACS will illuminate the POS BYP pushbutton.

C. This is correct because depressing the POS BYP pushbutton will illuminate the green indicating light adjacent to all the control rods bypassed in the RACS.

D. This is incorrect because depressing the SUBST POS pushbutton on P-680 will only indicate those rods that have substitute data entered, and not those that are bypassed.

Technical References:
GLP-OPS-C1102.03, page 29 of 50.

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the effect that a loss or malfunction of the Control rod drive system (plant-specific) will have on the RECIRCULATION SYSTEM (CFR: 41.7 / 45.7).	Tier #	2
	Group #	2
	K/A #	202001 K6.05
	Rating	2.7

Question 57

While operating at 100% power, the alarm CRD PMP SUCT BKW FLTR DP HI is received.

Sustained operations with this condition can affect the Recirculation System by:

A. Limiting the recirculation Flow Control Valve operation due to high HPU oil temperature

B. Lowering the cooling capacity of the recirculation pump motor oil heat exchangers

C. Lowering the cooling capacity of the recirculation pump shaft seal heat exchangers

D. Degrading the integrity of the recirculation pump shaft seals

Answer: D

Explanation:

A. This is incorrect because air provides cooling to HPU oil coolers.

B. This is incorrect because CCW provides cooling to this component.

C. This is incorrect based on the Explanation for Answer B.

D. This is correct because the CRD Hydraulic system provides cooling to the shaft seals in the recirculation pumps.

Technical References:

GLP-OPS-B3300, GLP-OPS-C111A, GFIG-OPS-B3300, GFIG-OPS-C111A

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including recorders and meters (CFR: 41.5 / 45.5).	Tier #	2
	Group #	2
	K/A #	216000 A1.01
	Rating	3.4

Question 58

The plant indications are as follows:

- Reactor coolant temperature is 120^0F
- Actual reactor vessel level is +35”
- Recirculation pumps A and B are running in slow speed

Given these conditions, the most accurate level indication would be the:

A. Narrow Range

B. Wide Range

C. Shutdown Range

D. Upset Range

Answer: C

Explanation:

A. This is incorrect because Narrow Range is calibrated for RPV pressure at 1025 psig, which result in error in the given indication.

B. This is incorrect because wide Range is calibrated for RPV pressure at 1025 psig with no recirculation flow, which results in error in the given indication.

C. This is correct because the conditions most closely match the calibrated conditions for this level instrumentation. The reactor coolant temperature at less than 200^0F implies that reactor pressure is 0 psig, which is one of the calibration conditions. With the available range being +0” to +400”, this is a level that would indicate on scale.

D. This is incorrect because the Upset Range is calibrated for RPV pressure at 1025 psig, which result in error in the given indication.

Technical References:

GLP-OPS-B2101

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5, 55.45.5	

Examination Outline Cross-Reference	Level	RO
Knowledge of RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE design feature(s) and/or interlocks which provide for pump minimum flow protection (CFR: 41.7).	Tier #	2
	Group #	2
	K/A #	226001 K4.05
	Rating	2.5

Question 59

RHR A is operating in Containment Spray (CS) Mode following an automatic initiation of Containment Spray.

A control room operator inadvertently places the control switch for E12-F004A, RHR PMP A SUCT FM SUPP POOL, in the CLOSE position.

As a result of this action, E12-F004A will:

A. close and RHR Pump A will trip.

B. close and then reopen with RHR Pump A continuing to run.

C. not close because of the CS signal.

D. not close because of the LOCA signal.

Answer: A.

Explanation:

A. This is correct because E12-F004A does not have any interlocks associated with it and the RHR pump will automatically trip if E12-F004A is not fully open.

B. This is incorrect because E12-F004A does not have any interlocks associated with it to reopen the valve.

C. This is incorrect because E12-F004A does not have any interlocks associated with containment sprays.

D. This is incorrect because E12-F004A has no LOCA signals associated with it.

Technical References:

GLP-OPS-E1200, GFIG-OPS-E1200

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of the operational implications of turbine inlet pressure vs. reactor pressure as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM: (CFR: 41.5 / 45.3).	Tier #	2
	Group #	2
	K/A #	241000 K5.04
	Rating	3.3

Question 60

A plant startup is in progress with the following:

- Reactor power is 8% on all APRMs
- Main turbine bypass valves are 20% open

A high reactivity worth control rod begins to drift out of the core.

No operator action has yet been taken.

Which of the following describes the <u>initial</u> response of reactor pressure and the main turbine bypass valves as the control rod continue to drift out of the core?

Reactor pressure:

A. lowers and the main turbine bypass valves begin to close.

B. rises and the main turbine bypass valves open more.

C. lowers and the main turbine bypass valves open more.

D. rises and the main turbine bypass valves begin to close.

Answer: B

Explanation:

As the rod drifts out, positive reactivity is added to the core, causing reactor power to rise and reactor pressure to rise. As reactor pressure rises, so does the pressure at the main steam equalizing header (for our purpose, we can say this is the same as turbine inlet steam pressure). At this point in the plant startup (no generator load), the Pressure Regulating System uses the turbine bypass valves to correct the rising steam pressure; it does this by opening the bypass valves more until pressure lowers back to setpoint.

For this reason, ‘B’ is correct and all distracters are incorrect.

Technical References:

GFIG-OPS-N3202

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.3	

Examination Outline Cross-Reference	Level	RO
Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the D. C. electrical distribution: (CFR: 41.2 to 41.9 / 45.7 to 45.8).	Tier #	2
	Group #	2
	K/A #	245000 K1.09
	Rating	2.7

Question 61

Which of the following DC buses power(s) the Main Turbine EHC Control (logic) System?

A. 11DB

B. 11DF

C. 11DH and 11DJ

D. 11DK and 11DL

Answer: C

Explanation:

A. This is incorrect because this Div II ESF DC has no relationship with the EHC system.

B. This is incorrect because this BOP 250 VDC bus powers only 250 VDC pumps.

C. This is correct per the DC System SOI (04-1-01-L11-1) load list.

D. This is incorrect. These two 125 VDC buses are connected in-series to make up the BOP 250 VDC 11DF bus; although each 125 VDC bus also powers other 125 VDC components, none of those components are associated with the EHC system.

Technical References:

GLP-OPS-N3202, GFIG-OPS-L1100

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2 to 55.41.9; 55.45.7 to 55.45.8	

Examination Outline Cross-Reference	Level	RO
Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including pump starts (CFR: 41.7 / 45.7).	Tier #	2
	Group #	2
	K/A #	256000 A3.02
	Rating	3.0

Question 62

The plant is in Mode 2 and:

- Condensate Pumps A and B are operating
- Condensate Booster Pumps A and B have recently been started
- Condensate Booster Pump suction pressure is 200 psig
- Condensate Booster Pump discharge pressure is 575 psig
- Condensate Booster Pump A discharge valve is partially open and the discharge valve for Pump B is full open
- Condensate Booster Pump A lube oil low pressure annunciator goes into alarm and seals in

Based on these conditions, the operator would expect Condensate Booster Pump A to trip due to:

A. low pump suction pressure for 15 seconds.

B. the pump's discharge valve being partially open for 15 seconds.

C. low discharge pressure.

D. low lube oil pressure condition for 15 seconds causing the suction valve to close.

Answer: D

Explanation:

A. This is incorrect because the low pump suction pressure automatic trip set point is 44 psig with a 15 second time delay. Current pump suction pressure is well above this set point.

B. This is incorrect because the automatic pump trip associated with the discharge valve is activated if the valve is fully closed for 15 seconds.

C. This answer is incorrect because this is the normal discharge pressure.

D. This answer is correct because the sustained low lube oil pressure (< 3 psig) for 15

seconds causes an automatic closure of the pump suction valve. If the pump's suction valve is not fully open, the associated condensate booster pump trips automatically.

Technical References:

GLP-OPS-N1900, GFIG-OPS-N1900, E-1148 Sheets 3 & 4

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.45.7	

Examination Outline Cross-Reference	Level	RO
Ability to (a) predict the impacts of following on the REACTOR FEEDWATER SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of Extraction Steam (CFR: 41.5 / 45.6) (A2.04)	Tier #	2
	Group #	2
	K/A #	259001 A2.04
	Rating	3.3

Question 63

Use your provided references to answer this question.

Following a refueling outage, the following conditions exist:

- Reactor power is 21%
- Extraction steam to Feedwater Heaters 5A and 6A is isolated for completion of outage work

A loss of extraction steam to Feedwater Heater 6B occurs.

After the transient, feedwater inlet temperature to the reactor stabilizes at 260°F and reactor power stabilizes at 26%.

According to Procedure 05-1-02-V-5, Loss of Feedwater Heating, the:

A. MCPR limits are no longer valid and the crew should take the generator off-line.

B. LHGR limits are no longer valid and the crew should scram the reactor.

C. MCPR limits are no longer valid and the crew should scram the reactor.

D. LHGR limits are no longer valid and the crew should take the generator off-line.

Answer: A

Explanation:

A. This is correct because with feedwater temperature 260 degrees and power 26 percent, the reactor is in Region III of the Feedwater Temperature versus Power map. This requires the crew to take the generator off-line per step 3.2.3 of the referenced procedure. According to step 1.4 of the same procedure, the MCPR limits are no longer valid.
B. This is incorrect because scramming the reactor is not required unless Region IV of the Feedwater Temperature versus Power map is entered.
C. This is incorrect because scramming the reactor is not required unless Region IV of the Feedwater Temperature versus Power map is entered.

D. According to step 1.4 of the same procedure, the MCPR limits are no longer valid.

Technical References: Procedure 05-1-02-v-5, Loss of Feedwater Heating

References to be provided to applicants during exam:

Attachment I of Procedure 05-1-02-v-5, Loss of Feedwater Heating

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for redundancy (CFR: 41.7).	Tier #	2
	Group #	2
	K/A #	295010 K4.01
	Rating	2.7

Question 64

The plant is operating at rated power when the high voltage power supply fails on Offgas Post-Treatment radiation monitor Channel A.

What is the response of the Offgas System?

A. The system auto isolates.

B. The system shifts to bypass mode.

C. The system shifts out of bypass mode if it was in bypass mode at the time.

D. The system continues to operate in its normal lineup.

Answer: D

Explanation:

It takes both Post-Treatment rad monitors to generate an Offgas system isolation; in any combination of INOP or HI-HI-HI (upscale). Stem indicate an INOP condition on only one channel. Therefore, the system remains in its normal lineup; thus, 'D' is correct.

Technical References:

GLP-OPS-D1721, GLP-OPS-N6465

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Fire Protection: Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on personnel protection: (CFR: 41.7 / 45.4) (K3.02)	Tier #	1
	Group #	2
	K/A #	286000 AK3.02
	Rating	3.2

Question 65

A fire has been reported in the Div I Switchgear Room on Control Building elevation 111'.

The Fire Brigade has been dispatched and has determined that the CO_2 System has failed to automatically initiate due to a trip of the control power breaker; the breaker will not reset.

What actions must be taken to manually initiate CO_2 and put the fire out?

A. Place the Manual Release Lever at the 10-Ton CO_2 tank to the OPEN position and rotate the Local Manual Actuation Arm to the OPEN position.

B. Rotate the Local Manual Actuation Arm to the OPEN position, ONLY.

C. Place the Manual Release Lever at the 10-Ton CO_2 tank to the OPEN position and depress the local CO_2 Manual Initiation Push Button.

D. Depress the local CO_2 Manual Initiation Push Button, ONLY.

Answer: A

Explanation:

A. This answer is correct because the Manual Release Lever at the 10 Ton CO2 tank must be rotated to the OPEN position and the Local Manual Actuation Arm must be rotated to the OPEN position to initiate CO2 with loss of control power.

B. This answer is incorrect because the Manual Release Lever at the 10 Ton CO2 tank must be rotated to the OPEN position and the Local Manual Actuation Arm must be rotated to the OPEN position to initiate CO2 with loss of control power.

C. This answer is incorrect because the local CO2 Manual Initiation Push Button will not work with loss of control power. Also, Manual Release Lever at the 10 Ton CO2 tank is not required if Manual Initiation Push Button functional.

D. This answer is incorrect because the local CO2 Manual Initiation Push Button will not work with loss of control power.

Technical References:

GLP-OPS-P6400, Page 32 of 48; 04-1-01-P64-3 SOI

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Examination Outline Cross-Reference	Level	RO
Knowledge of facility requirements for controlling vital/controlled access. (CFR: 41.10 / 43.5 / 45.9 / 45.10)	**Tier #**	Administrative
	Group #	
	K/A #	Generic - 2.1.13
	Rating	2.5

Question 66

While transiting between spaces in the plant, an operator uses his/her card key to attempt access into a vital area.

He/she swipes the card key, then a red light is displayed with audible beeps heard.

The next action of the operator is to:

A. Swipe the card key again and open the door when the audible beeps are heard.

B. Open the door and enter the vital area.

C. Contact security for assistance.

D. Swipe the card key again and open the door when a green light is displayed.

Answer: D

Explanation:

A. The audible beeps, along with the red light, indicate that access has been rejected. The vital area door will not allow access.

B. With the first swiping of the card key, the indications are that access has been rejected. Therefore, the system will not allow one to open the door.

C. Per procedure 01-S-11-10, the card key needs to be swiped a second time upon initial access rejection prior to contacting security for assistance.

D. With the initial access rejection, the individual needs to swipe the card key another time. If a green light is displayed, the door to the vital area can be opened.

Technical References:

Procedure 01-S-11-10, Sections 6.11.2 and 6.12

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.9, 55.45.10	

Examination Outline Cross-Reference	Level	RO
Ability to make accurate, clear, and concise verbal reports. (CFR: 41.10 / 45.12 / 45.13)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.1.17
	Rating	3.9

Question 67

After the ACRO inserts a manual scram with the Mode Switch, the following indications are observed:

- SCRAM AIR HDR PRESS LO annunciator is sealed-in
- RPS Group lights (white lights) are all extinguished
- Multiple control rods are still at their pre-scram positions
- Reactor power is 18%
- MSIVs are closed
- Feedwater pumps have tripped on high level

The ACRO is expected to give the following scram report:

A. "Mode switch is in shutdown, we have a hydraulic lock ATWS, bypass valves are available, feedwater is recoverable."

B. "We have a hydraulic lock ATWS, reactor power is 18 percent, mode switch is in shutdown, bypass valves are not available, feedwater is recoverable."

C. "Mode switch is in shutdown, we have an electrical ATWS, reactor power is 18 percent, bypass valves are not available, feedwater is available."

D. "We have an electrical ATWS, reactor power is 18 percent, bypass valves are available, feedwater is recoverable."

Answer: B

Explanation:

Stem conditions indicate the following: 1) a hydraulic lock ATWS, not an electrical one (i.e., RPS did function per design); 2) bypass valves are not available (because the MSIVs are closed); and 3) having tripped on high level, the feedwater pumps are recoverable (though not currently "available").

Answer 'B' mimics the currently-approved "Operating Hardcard" (kept at P680) for this required scram report.

For the above reasons, all 3 distracters are incorrect.

Technical References:

Procedure 01-S-06-14, GFIG-OPS-E1200, GLP-OPS-E1200

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)	**Tier #**	Administrative
	Group #	
	K/A #	Generic - 2.1.39
	Rating	3.6

Question 68

Xenon is still changing while a shift turnover is in progress.

The offgoing CRS directs the offgoing ACRO to lower power by 1 MWe using Recirc.

How should this evolution transpire?

A. The oncoming ACRO should assume the ACRO duties and make the power change; no Peer Check is required.

B. The offgoing ACRO should make the power change prior to completing the turnover; a Peer Check is required.

C. The offgoing ACRO should maintain the ACRO duties; the oncoming ACRO should make the power change; a Peer Check is required.

D. The offgoing ACRO should make the power change prior to completing the turnover; no Peer Check is required.

Answer: B

Explanation:

A. The Shift Turnover practices stated in the Conduct of Operations procedure (Section 5.16) does allow for turning over the shift to the oncoming operator if changes in the equipment are understood by the oncoming operator. However, in cases when reactivity changes (Section 5.4) are directed, they are to be avoided at shift turnover. Taking controls of a directed reactivity change at shift turnover is not the most conservative choice.

B. Since a directed reactivity change needs to be taken care of near shift turnover time, it is best to allow the current Reactor Operator to make the change prior to relieving his/her position. The current Reactor Operator has the knowledge of the situation from his/her previous shift, and would be most capable of completing the change.

C. While this could be conducted this way, the offgoing Reactor Operator may have other duties to attend to that may limit their ability to assist in the reactivity change. Also,

related to Answer B., the current Reactor Operator is most capable of making the change based on knowledge of the situation during the past shift.

D. The oncoming Reactor Operator, when he/she assumes the controls, is responsible for changes in reactivity at the controls. Allowing someone else to make the changes is not consistent with responsibility of the on-shift Reactor Operator.

Technical References:

Procedure EN-OP-115, Sections 5.4 and 5.16

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. (CFR: 41.10 / 43.5 / 45.13)	Tier #	
	Group #	
	K/A #	Generic – 2.2.18
	Rating	2.6

Question 69

The plant is performing a risk management review of tasks planned for an upcoming refueling outage.

A review of the requirements at all given times during an outage is required for emergency core cooling systems.

Additional systems that require this type of review are:

A. Plant Chilled Water and Fuel Pool Cooling & Cleanup systems

B. Shutdown Cooling and Fuel Handling Area Ventilation systems

C. Fuel Transfer and Fuel Pool Cooling & Cleanup systems

D. Shutdown Cooling and Fuel Pool Cooling & Cleanup systems

Answer: D

Explanation:

A. This is incorrect because one of the systems listed is not part of those stated in Procedure 01-S-18-6, Section 6.4.2.b.

B. See Explanation for Answer A.

C. See Explanation for Answer A.

D. This is correct based on additional risk management requirements stated in Procedure 01-S-18-6, Section 6.4.2.b. Three system classifications must include a review of requirements at any given time during the outage [Sub-item b)].

Technical References:

GLP-SM-MRP, Procedure 01-S-18-6

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.2.37
	Rating	3.6

Question 70

Per Technical Specification 3.5.1, ECCS – Operating, the HPCS pump must produce a flow rate of ≥ 7115 gpm at a Total Developed Head of _____.

A. ≥ 125 psid

B. ≥ 290 psid

C. ≥ 340 psid

D. ≥ 445 psid

Answer: D

Explanation:

A. This is incorrect; this is the Tech Spec value for the LPCI pumps.

B. This is incorrect; this is the Tech Spec value for the LPCS pump.

C. This is incorrect; this has face plausibility because it sounds like the generally accepted value for the shutoff head of the LPCI pumps (~340 psig).

D. This is correct per Tech Spec SR 3.5.1.4. .

Technical References:

Technical Specification, LCO 3.5.1

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7, 55.43.5, 55.45.12	

Examination Outline Cross-Reference	Level	RO
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.2.44
	Rating	3.6

Question 71

The plant is operating at rated power.

Control room operators are performing the Reactor Feed Pump Emergency Governor Exercise – Monthly, 04-1-03-N21-5.

While the TEST LOCKOUT pushbutton for RFPT 'A' is being held depressed, RFPT 'A' experiences an actual over-speed condition.

Which of the following describes the automatic response of RFPT 'A'?

A. Trips as a result of the electrical trip solenoid.

B. Backs down and stabilizes at a lower speed.

C. Ramps up due to the overspeed condition and locks at a high speed stop.

D. Does not trip until the TEST LOCKOUT pushbutton is released; then it trips.

Answer: A

Explanation:

A. This answer is correct because the test blocks the mechanical device trip but leaves the electrical trip system active and there for only the electrical trip is active during the test.

B. This answer is incorrect because there is no governor signal that backs the RFPT speed down on an overspeed condition.

C. This answer is incorrect because even though it will ramp up, there is no protective function short of the electrical overspeed trip device.

D. This answer is incorrect because the test blocks the mechanical device trip but leaves the electrical trip system active and there for only the electrical trip is active during the test.

Technical References:

GLP-OPS-N2100.09, Page 24 of 58.

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.4	

Examination Outline Cross-Reference	Level	RO
Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.3.4
	Rating	3.2

Question 72

Per 10-S-01-17, Emergency Personnel Exposure Control, which of the following is the maximum acceptable dose limit extension (TEDE) for an emergency worker?

A. 50 Rem to save a life (to be absorbed by a volunteer emergency worker aware of the risks)

B. 30 Rem to protect large populations (to be absorbed by an involuntary emergency worker)

C. 20 Rem to protect valuable property

D. 15 Rem to protect valuable property

Answer: A

Explanation:
Refer to 10-S-01-17, section 6.1 Table. Only 'A' is correct.

Technical References:

Procedure EN-RP-105; 10-S-01-17

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.12, 55.43.4, 55.45.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.3.14
	Rating	3.4

Question 73

Compared to a typical sample, a coolant sample indicative of a fuel leak would contain higher than normal amounts of:

A. Chlorine-38

B. Plutonium-239

C. Manganese-56

D. Cobalt-60

Answer: B

Explanation:

A. This is incorrect because this is a coolant activation product with a half-life of 37 minutes. Finding coolant activation products in a coolant sample is expected during normal plant operations.

B. This is correct because the isotope is a transuranic. Except in trace quantities, these should not be found in the reactor coolant unless fuel failures are present.

C. This is incorrect because it is an activated corrosion product that is typically found in a coolant sample during normal plant operations.

D. See Explanation for Answer C.

Technical References:

GLP-OPS-MCD06

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.12, 55.43.4, 55.45.10	

Examination Outline Cross-Reference	Level	RO
Knowledge of crew roles and responsibilities during EOP usage. (CFR: 41.10 / 45.12)	**Tier #**	Admin
	Group #	
	K/A #	Generic - 2.4.13
	Rating	4.0

Question 74

The plant is operating at rated power when the following occurs:

- Drywell pressure reaches 1.25 psig
- Reactor water level lowers to +11”
- No automatic scram has occurred

Per EN-OP-115, Conduct of Operations, how should the ACRO respond?

A. Notify the CRS of the condition and await direction.

B. Notify the CRS of the condition, recommend a manual scram, and await direction to insert the manual scram.

C. Initiate a manual scram and announce the action to the CRS.

D. Allow time for another RO to validate the condition, then initiate a manual scram and announce the action to the CRS.

Answer: C

Explanation:

A. This is incorrect; see explanation for ‘C’.

B. This is incorrect; see explanation for ‘C’.

C. This is correct because the situation involves a situation that requires a manual scram of the reactor immediately. Procedure EN-OP-115, Section 5.4, Step [24] says that emergency reactivity changes require authorization of the SRO (Control Room Supervisor) with Control Room Command and Control, unless an immediate manual scram is required by Subsection 5.1. One of the conditions in Subsection 5.1 that requires an immediate manual reactor scram is when operating parameters exceed any of the reactor protection set points and an automatic shutdown does not occur. With the given indications, two RPS scram set points are exceeded (Reactor Low Level of 11.4” and High Drywell Pressure of 1.23 psig).

D. This is incorrect; see explanation for ‘C’.

Technical References:

GFIG-OPS-C1102, GLP-OPS-C1102, GLP-OPS-C7100, Procedure EN-OP-115

References to be provided to applicants during exam:

None

Learning Objective: To be determined

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.45.12	

Examination Outline Cross-Reference	Level	RO
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)	**Tier #**	3
	Group #	
	K/A #	2.4.21
	Rating	4.0

Question 75

The reactor is operating in Mode 1 when a RECIRC PMP B SEAL STG FLO HI/LO alarm is received.

The following indications exist for Reactor Recirc Pump B:

- Seal staging flow is high (computer point)
- #1 Seal cavity pressure is 970 psig
- #2 Seal cavity pressure is 800 psig

Based on these indications, Reactor Recirc Pump B has a:

A. Degraded #1 seal.

B. Degraded #2 seal.

C. Plugged seal staging flow orifice.

D. Plugged pressure breakdown orifice.

Answer: A

Explanation:

A. This is the correct because as the #1 seal degrades, its pressure drop goes down, resulting in the #2 seal cavity pressure rising toward the #1 seal cavity pressure. Staging flow rises as a result of the degraded #1 seal; hence, the annunciator cited in the stem; in this case, for HI FLO.

B. This answer is incorrect because a degraded #2 seal would result in its seal cavity pressure lowering (towards Drywell atmospheric pressure), not rising toward #1 seal pressure.

C. This is incorrect because a plugged seal staging orifice will cause a low seal staging flow, not a high flow.

D. This is incorrect because a plugged pressure breakdown orifice will cause a low seal staging flow, and cause #2 seal pressure to lower, not rise.

Technical References:

References to be provided to applicants during exam:

None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret the jet pump operability as it applies to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295001 AA2.05
	Rating	3.4

Question 76

Use your provided references to answer this question.

With the plant operating at rated power, the daily Jet Pump Functional Test is completed at 2100 on 1/15/09, with the following results.

- Jet Pump #13 indicated flow deviates from established patterns by -23%
- All other Jet Pump indicated flows deviate from established patterns by less than 5%
- Recirculation Loop 'A' Drive flow is 38.9 Kgpm
- Recirculation Loop 'B' Drive flow is 39.5 Kgpm
- Recirculation loop drive flow versus flow control valve position differs by -2% for the 'A' Loop and -12% for the 'B' Loop.
- Total recirculation loop drive flow versus total core flow differs by 1%

Which of the following describes the Technical Specifications required action?

A. Shut down one recirculation loop no later than 2300 on 1/15/09.

B. Place the plant in Mode 3 no later than 0900 on 1/16/09.

C. Perform a special test per LCO 3.0.7, Special Operations LCOs.

D. No Technical Specification action is required.

Answer: B

Explanation:

A. This answer is incorrect because this LCO is only applicable if there is a jet pump flow mismatch per TS 3.4.1. In this case the flows are matched so there is no entry into this LCO.

B. This answer is correct because per TS Surveillance 3.4.3.1, since criteria a and c of SR 3.4.3.1 is not met, With 2 of the 3 criteria not met, the Jet Pump #13 is inoperable.

C. This answer is incorrect because it only applies to LCOs that are associated with

special operations in section 3.10 of TS.

D. Since 2 of the 3 Jet Pump criteria not met per SR 3.4.3.1, LCO action is required.

Technical References:

LGGNS Technical Specification LCOs 3.4.1 and 3.4.3

References to be provided to applicants during exam:

GGNS Technical Specifications 3.4.1 and 3.4.3

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.2, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Control Room Abandonment - Knowledge of abnormal condition procedures (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295016 G – 2.4.11
	Rating	4.2

Question 77

Use your provided references to answer this question.

Due to a fire in the control room, the control room is evacuated at 0930 hours.

The ONEP for Shutdown from the Remote Shutdown Panel is being implemented.

It takes 20 minutes to get to the Remote Shutdown Panel; plant control is established at 0950 hours.

Based on this sequence of events, the SRO should declare a(n):

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. General Emergency.

Answer: C

Explanation:

A. This answer is incorrect based on the Explanation for Answer C.

B. This answer is incorrect based on the Explanation for Answer C.

C. This answer is correct because a Control Room evacuation with no plant control established within 15 minutes results in a Site Area Emergency per Emergency Plan Procedure 10-S-01-1, Attachment I, Emergency Classification HS3. There is a declaration of an Alert for the Fire Event (Procedure 10-S-01-1, Attachment I, Emergency Classification HA4, but the higher of the two classification levels is dominant.

D. This answer is incorrect based on the Explanation for Answer C.

Technical References:

Lesson Plan GLP-OPS-C6100, Procedure 05-1-02-II-1, ONEP 10-S-01-1

References to be provided to applicants during exam:

EAL flowcharts

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
High Reactor Pressure - Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12).	Tier #	1
	Group #	1
	K/A #	295025 G – 2.1.20
	Rating	4.6

Question 78

Following a reactor scram, the following conditions exist:

- All control rods are fully inserted
- MSIVs are closed
- Reactor pressure is 1070 psig with one SRV cycling
- RPV water level cannot be determined
- RCIC and HPCS are unavailable
- All radiation levels are normal
- Suppression pool level is 10 feet and stable

The CRS should direct the crew to:

A. Transition to Steam Cooling.

B. Control RPV pressure with SRVs or RWCU.

C. Depressurize the RPV by opening 8 ADS/SRVs.

D. Defeat the necessary isolation interlocks and depressurize the RPV using Alternate Depressurization Systems.

Answer: D

Explanation:

A. This answer is incorrect because it suggests an action that is found in EP-2; with RPV water level unknown, the crew should be in EP-5 (RPV Flooding) where no such action is found.

B. This answer is incorrect because SRVs cannot be used for RPV pressure control due to low SP level (<10.5').

C. This answer is incorrect because SRVs cannot be used with SP level <10.5'.

D. This answer is correct because with SP level <10.5', and HPCS unavailable, EP-5 directs us to depressurize using Table 3 Alternate Depressurization Systems.

Technical References:

Emergency Procedures EP-2 and EP-5

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret reactor pressure as it applies to SUPPRESSION POOL HIGH WATER TEMPERATURE (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295026 EA2.03
	Rating	4.0

Question 79

The reactor was scrammed due to a stuck open SRV.

Currently:

- RPV pressure is 1000 psig and slowly lowering
- RPV level is +30” and stable
- Suppression pool temperature is 132°F and rising
- The Heat Capacity Temperature Limit (HCTL) is being approached, but exceeding the limit is <u>not</u> imminent

Based on these conditions, the CRS should:

A. Open all ADS valves and emergency depressurize the RPV per EP-3, Containment Control.

B. Rapidly depressurize the RPV using alternate depressurization systems per EP-2, RPV Control.

C. Maintain RPV pressure less than HCTL using the pressure leg of EP-2, RPV Control.

D. Maintain RPV pressure less than the Pressure Suppression Pressure (PSP) using alternate pressure control systems per EP-3, Containment Control.

Answer: C

Explanation:

A. This answer is incorrect because EP-3 does not require an ED unless suppression pool temperature and RPV pressure cannot be maintained in the safe zone.

B. This answer is incorrect because while there is a need to reduce RPV pressure, it must be done within the cooldown limitations.

C. This answer is correct because step EP-3 requires the HCTL be maintained in the safe

zone and RPV pressure is being controlled by EP-2.

D. This answer is incorrect because RPV pressure is not one of the variables for the PSP.

Technical References:

Procedure 05-S-01-SAP-01, Procedure 05-S-01-EP-01, Procedure 05-S-01-EP-02, Procedure 05-S-01-EP-03

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
High Drywell Temperature - Knowledge of EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295028 G – 2.4.6
	Rating	4.7

Question 80

Following a reactor scram, the following conditions exist:

- Reactor power is 18% and slowly trending down
- MSIVs are open
- Drywell pressure is 1.40 psig
- Drywell temperature is 320°F and rising
- Suppression pool level is 19’ 8” and slowly rising

The CRS should order the crew to depressurize the RPV by implementing:

A. EP-2 and opening 8 ADS/SRVs.

B. EP-2 and opening the main turbine bypass valves.

C. EP-2A and opening 8 ADS/SRVs.

D. EP-2A and opening the main turbine bypass valves.

Answer: C

Explanation:

A. This answer is incorrect because EP-2 is for a non-ATWS transient.

B. This answer is incorrect because EP-2 is for a non-ATWS transient.

C. This answer is correct because with drywell temperature approaching the 330°F limit, an emergency depressurization is required by EP-3. With an ATWS in progress, EP-2A is the EP from which the ED will be implemented, using 8 ADS/SRVs.

D. This answer is incorrect because EP-2A requires 8 ADS/SRVs be opened.

Technical References:

Emergency Procedures EP-1, EP-2A, and EP-3

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret adequate core cooling as it applies to REACTOR LOW WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	1
	K/A #	295031 EA2.04
	Rating	4.8

Question 81

A LOCA and reactor scram have occurred with the following:

- One control rod is stuck at position 40
- All other control rods are fully inserted
- RPV pressure is 750 psig
- All high pressure injection capability has been lost
- RPV level is -150” and lowering slowly
- Off-Site power has been lost

The CRS should be implementing:

A. EP-2A, ATWS RPV Control.

B. The Alternate Level Control leg of EP-2.

C. The Emergency Depressurization leg of EP-2.

D. EP-5, RPV Flooding.

Answer: B

Explanation:

A. This answer is incorrect because the Override criteria to enter EP-2A are not met. With only one stuck rod, no ATWS condition exists. Therefore, there is one rod that is withdrawn past the indicated position, and it can be determined that the reactor will remain shutdown in all cases without boron (02-S-1-40, Attachment IV, Page 5) .

B. This answer is correct because with RPV level and injection source availability, the RPV level cannot be maintained above -160”. The Alternate Level Control leg is required.

C. This answer is incorrect because additional steps need to be taken with Alternate Level Control (EP-2) before entering Emergency Depressurization. Additional means of restoring level need to be attempted prior to RPV level decreasing below -191”.

D. This answer is incorrect because the RPV level is known and none of the conditions provided call the validity of level indication into question (Caution #1 in EP-1 or Attachment III in Emergency Procedure 02-S-1-40).with only one stuck rod, no ATWS condition exists; therefore, with an unknown RPV level, the crew would enter EP-5 rather than EP-5A.

Technical References:

Emergency Procedures EP-2, EP-5, and EP-5A

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10, 55.43.5, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Plant Fire On Site - Knowledge of the emergency plan. (CFR: 43.5)	Tier #	1
	Group #	1
	K/A #	600000 G – 2.4.29
	Rating	3.9

Question 82

Use your provided references to answer this question.

With the plant in MODE 1, an Alert would have to be declared if:

A. A fire in the Fire Water Pump House took 30 minutes to extinguish and resulted in a loss of Diesel Fire Pump 'A'.

B. A fire in the snubber test facility trash can (185' Aux Bldg) burned for 22 minutes and burned trash and operating manuals.

C. A barrel of oily rags burned for 25 minutes in the Division I Emergency Diesel Generator room and the fire damaged the EDG control panel.

D. A fire in the maintenance welding shop burned for 20 minutes and resulted in 4 injured mechanics being transported to the hospital.

Answer: C

Explanation:

A. This answer is incorrect because this is not a Table H2 or H3 structure. Therefore no EAL is met.

B. This answer is incorrect because no equipment or structure damaged. Therefore this would be an unusual event.

C. This answer is correct because there was a fire in a Table H2 structure damaged a safety system required for safe shutdown of the reactor.

D. This answer is incorrect because of the maintenance shop is not a structure identified as applicable for an Alert classification and the 4 injuries do not affect the emergency plan declaration. Therefore, no EAL is required.

Technical References:

Emergency Plan Procedure 10-S-01-1

References to be provided to applicants during exam:

EAL flowcharts

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.6, 55.45.4	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret the drywell temperature as it applies to HIGH DRYWELL TEMPERATURE (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	295012 AA2.01
	Rating	3.9

Question 83

The plant is operating at rated power when a loss of drywell cooling occurs.

Drywell average air temperature has reached 140°F and continues to slowly rise.

Per the Bases for Technical Specification 3.6.5.5, Drywell Average Air Temperature, if this condition is not corrected then a design basis LOCA could result in the design:

A. heat capacity of the suppression pool being exceeded.

B. drywell temperature limit of 330°F being exceeded.

C. fuel zone level instrumentation temperature limit being exceeded.

D. limit of 2% hydrogen in the drywell being exceeded.

Answer: B

Explanation:

A. This answer is incorrect because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

B. This answer is correct because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

C. This answer is incorrect because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

D. This answer is incorrect because the Technical Specification Bases document for high drywell temperature states the design drywell temperature of 330 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 135 degrees.

Technical References:

GGNS Technical Specifications Bases for technical specification 3.6.5.5.

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to determine and/or interpret reactor pressure as it applies to HIGH SUPPRESSION POOL WATER LEVEL (CFR: 41.10 / 43.5 / 45.13).	Tier #	1
	Group #	2
	K/A #	295029 EA2.02
	Rating	3.6

Question 84

The plant is operating at rated power.

A non-isolable leak has resulted in suppression pool level being above the upper limit established by the Technical Specifications and the level is still slowly rising.

Per the Bases for Technical Specification 3.6.2.2, Suppression Pool Water Level, with suppression pool level above the upper limit:

A. The peak drywell design pressure may be exceeded during a design basis LOCA.

B. There could be excessive clearing loads on the S/RV if the RPV is at rated pressure.

C. RCIC may trip on high exhaust back-pressure.

D. The peak containment design pressure may be exceeded during a design basis LOCA.

Answer: B

Explanation:

A. This answer is incorrect because the bases document does not state the drywell design pressure could be exceeded with a high suppression pool level but is credible because drywell pressure would be higher given a DBA LOCA and a higher SP water level.

B. This answer is correct because the referenced bases document states the upper limit is based, in part on precluding excessive dynamic loading on the S/RV.

C. This answer is incorrect because the bases document does not state the RCIC turbine may trip with a high suppression pool level but is credible because with a higher SP water level, RCIC back pressure would be higher.

D. This answer is incorrect because the bases document states the containment design pressure would not be exceeded with a high suppression pool level.

Technical References:

Technical Specification Bases for TS 3.6.2.2

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
High Secondary Containment Area Radiation Levels - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9).	Tier #	1
	Group #	2
	K/A #	295033 G 2.3.15
	Rating	3.1

Question 85

Use your provided references to answer this question.

The plant is operating at rated power.

At 0700 on 3/1/2009, while reviewing the logs, the CRS discovers the following:

- The last time that a Channel Check was performed on Fuel Handling Area Ventilation radiation monitor D17-K617A was 24 hours ago

Per Technical Specifications, the CRS is required to:

A. Immediately declare D17-K617A INOPERABLE and enter LCO 3.3.6.2.

B. Perform the surveillance no later than 1900 on 3/1/2009.

C. Perform the surveillance no later than 0700 on 3/2/2009.

D. Immediately declare D17-K617A INOPERABLE and enter LCO 3.3.6.2 and perform the surveillance no later than 1900 on 3/1/2009.

Answer: C

Explanation:

A. This answer is incorrect because TS SR 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required.

B. This answer is incorrect because TS SR 3.0.3 states the surveillance must be performed within the periodicity or 24 hours; whichever is longer.

C. This answer is correct because TS SR 3.0.3 allows up to 24 hours to complete a missed surveillance.

D. This answer is incorrect because TS SR 3.0.3 states if a surveillance is missed then immediate entry into the LCO is not required.

Technical References:

Lesson Plan GLP-OPS-D1721 (lesson plan and figures), Lesson Plan GLP-OPS-T4801, GGNS Technical Specifications Section 3.3.6.2 (LCO and Bases)

References to be provided to applicants during exam:

GGNS Technical Specification LCO 3.3.6.2.

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.12, 55.43.4, 55.45.9	

Examination Outline Cross-Reference	Level	SRO
Shutdown Cooling - Knowledge of procedures and limitations involved in core alterations. (CFR: 41.10 / 43.6 / 45.7).	Tier #	2
	Group #	1
	K/A #	205000 G 2.1.36
	Rating	4.1

Question 86

Use your provided references to answer this question.

The plant is in MODE 5 with the following:

- Reactor water level is 23 feet above the RPV flange
- Irradiated fuel is being moved within the reactor vessel
- One qualified circuit between the off-site transmission network and Divisions 1, 2, and 3 electrical power distribution systems are OPERABLE
- DG 11 is tagged out for repairs
- RHR 'B' is operating in Shutdown Cooling
- ADHRS is OPERABLE
- Reactor Recirc and RWCU are out of service for maintenance

If the RHR 'B' pump trips, then the immediate (≤ 1 hour) required actions per Technical Specification 3.9.8 are:

A. 3.9.8.A.1, ONLY.

B. 3.9.8.C.1, ONLY.

C. 3.9.8.C.1 and 3.9.8.C.2, ONLY.

D. 3.9.8.A.1 and 3.9.8.C.1 and 3.9.8.C.2.

Answer: D

Explanation:

A. This answer is incorrect because it list only part of the required actions, implying that one RHR cooling subsystem is in operation when it is not.

B. This answer is incorrect because it assumes RHR A is OPERABLE and it is not (per the bases document). Also, there are two actions for this condition that are tied by an AND logical connector making both actions required.

C. This answer is incorrect because it assumes RHR A is OPERABLE and it is not (per the bases document).

D. This answer is correct because it addresses the conditions of no operable RHR pumps and no decay heat removal subsystem in operation.

Technical References:

Procedure 05-1-02-III-1; Lesson Plans GLP-OPS-P7500, GLP-OPS-R2100; GGNS Technical Specifications LCOs 3.9.8 and 3.9.9

References to be provided to applicants during exam:

GGNS Technical Specifications 3.9.8

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Source Range Monitor – Ability to diagnose and recognize trends in a timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12).	Tier #	2
	Group #	1
	K/A #	215004 G 2.4.47
	Rating	4.0

Question 87

Use your provided references to answer this question.

At 0800, a reactor startup is in progress with the following:

- All SRMs are indicating 2 x 10^2 CPS
- All IRMs are on Range 2

At 0800, SRM 'A' is declared inoperable.

At 0810, the following conditions exist:

- All SRMs are declared inoperable based on a common mode failure
- All IRMs are on Range 2

Which of the following describes the **EARLIEST** action required by Technical Specification 3.3.1.2, Source Range Monitor Instrumentation?

A. Restore all SRMs to an OPERABLE status no later than 1200.

B. Suspend rod withdrawal at 0810.

C. Begin implementing the actions as required by Tech Spec 3.0.3 at 0810.

D. Fully insert all insertable control rods <u>and</u> place the Mode Switch in SHUTDOWN no later than 0910.

Answer: B

Explanation:

A. This answer is incorrect because with only a single inoperable SRM, the minimum OPERABLE channels specified in Table 3.3.1.2-1 is still being satisfied; there is no requirement to enter TS 3.3.1.2, yet.

B. This answer is correct because now the Table 3.3.1.2-1 requirements are not being satisfied; therefore, TS 3.3.1.2 must be entered. Once entered, ACTION B.1 applies. Although ACTION A.1 also applies, B.1 is the EARLIEST of the two.

C. This answer is incorrect because the way that TS 3.3.1.2 is written, there is no combination of inoperable SRMs that can ever lead to an entry for TS 3.0.3.
D. This answer is incorrect because ACTION D.1 can only apply in Mode 3 or 4. During a reactor startup, the given stem conditions are only possible in Mode 2.

Technical References:

GGNS Technical Specification 3.3.1.2

References to be provided to applicants during exam:

Tech Spec 3.3.1.2

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
PCIS/Nuclear Steam Supply Shutoff - Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13).	Tier #	2
	Group #	1
	K/A #	223002 G - 2.2.39
	Rating	4.5

Question 88

Use your provided references to answer this question.

The plant is operating at rated power.

At 0700 on 8/1, the CRS is informed that the Containment/ Drywell ventilation exhaust radiation monitors, D17-K609A and D17-K609B, have failed the Surveillance Requirement SR 3.3.6.1.2, Channel Functional Test.

No operator action has yet been taken.

What is the Technical Specifications required action?

A. Isolate the affected penetration flowpath(s) no later than 0900 on 8/1.

B. Place both channels in trip no later than 0800 on 8/2.

C. Isolate the affected penetration flow path(s) no later than 0800 on 8/2.

D. Place one channel in trip no later than 0800 on 8/1.

Answer: A

Explanation:

A. This answer is correct because the isolation logic is 2-out-of-2 taken once logic and therefore, the isolation capability is lost. The CRS will have to restore isolation capability within 1 hour per TS 3.3.6.1.B or enter TS 3.3.6.1.C. TS 3.3.6.1.C requires the CRS immediately enter table 3.3.6.1-1 and per Condition F isolate the affected penetrations within one hour. Therefore the penetrations must be isolated no later than 0900 on 8/1.

B. This answer is incorrect because condition A requires the affected channels be placed in trip within 24 hours, or no later than 0700 on 8/2.

C. This answer is incorrect because the penetrations need to isolated no later than 0900 on 8/1 per the discussion in A above..

D. This answer is incorrect because the applicable LCO (A) provides 24 hours to place both channels in a tripped condition.

Technical References:

Lesson Plan GLP-OPS-M7101 (and associated Figures), Lesson Plan GLP-OPS-D1721 (and associated Figures), GGNS Technical Specifications (LCO 3.3.6.1)

References to be provided to applicants during exam:

GGNS Technical Specification 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation including applicable SRs and Table 3.3.6.1-1.

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7, 55.41.10, 55.43.2, 55.45.13	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High system pressure (plant-specific) (CFR: 41.5 / 45.6).	**Tier #**	2
	Group #	1
	K/A #	261000 A2.14
	Rating	3.2

Question 89

Use your provided references to answer this question.

At 1200 on 6/1, the plant is operating at rated power when operators commence the SGTS Train 'A' monthly surveillance.

At 1500, the SGTS FLTR TR A HEPA/CHAR ΔP HI alarm annunciates.

At 2300, while reviewing the completed surveillance, the CRS discovers an error and determines that the SGTS flow rate does not meet the surveillance Acceptance Criteria.

What is the Technical Specifications required action?

Declare SGTS 'A' inoperable effective at…

A. 1500 and perform SR 3.6.4.3.1 on SGTS 'B' no later than 0300 on 6/2.

B. 2300 and restore SGTS 'A' subsystem operability by 2300 on 6/8.

C. 1200 and restore SGTS 'A' subsystem operability by 1200 on 6/8.

D. 2300 and restore SGTS 'A' subsystem operability by 1500 on 6/8.

Answer: B

Explanation:

A. While it may be prudent to test the other train, this answer is incorrect because there is no TS requirement to perform an SR on the other train. Also, entry time is time of discovery.

B. This answer is correct because the entry time into an LCO is based on the time of discovery. Since this was at 2300, the inoperablility time would be 2300 and the TS requires it be returned to an operable status within 7 days.

C. This answer is incorrect because the entry time into an LCO is based on the time of

discovery. While the surveillance was started at 1200, the inoperability was not discovered until 2300 by the CRS. Therefore, the LCO entry time would be 2300.

D. This answer is incorrect because the restoration time was based on initial surveillance start time and not time of discovery.

Technical References:

Lesson Plan GLP-OPS-T4801, GGNS Technical Specifications Section 3.6.4.3

References to be provided to applicants during exam:

GGNS Technical Specification 3.6.4.3

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of Grounds on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations (CFR: 41.5 / 45.6).	Tier #	2
	Group #	1
	K/A #	263000 A2.01
	Rating	3.2

Question 90

Use your provided references to answer this question.

The plant is operating at rated power when it is discovered there are ground faults on the Div I battery due to faulty parts installed during a recent modification.

In response, at 0800 on 3/1, the Div I DC power subsystem is declared inoperable.

At 1200 on 3/1, the Div II DC power subsystem is declared inoperable for the same reason.

At 1600 on 3/1, the Div II DC power subsystem ground faults are corrected and it is declared OPERABLE.

However, there are no remaining replacement parts in stock for Div I.

If the Div I DC power subsystem is not returned to an OPERABLE status, the plant must be in MODE 3 no later than:

A. 2200 on 3/1.

B. 2300 on 3/1.

C. 0100 on 3/2.

D. 0200 on 3/2.

Answer: A

Explanation:

A. This answer is correct because with one division inoperable condition 3.8.4.C is entered. When the second division is declared inoperable at 1200, 3.0.3 is entered. When Div II is declared operable, 3.0.3 is exited leaving 3.8.4.C in effect. At 1000 3.8.4.E is entered requiring the unit to be in MODE 3 in 12 hours or at 2200.

B. This answer is incorrect based on the discussion for choice ‘A’. It would be correct if

the time was based on entering 3.0.3 at 1000.

C. This answer is incorrect based on the discussion for distractor A. This would be correct if based on entering 3.0.3 at 1200.

D. This answer is incorrect based on the discussion for distractor A. This would be correct if based on the second train being inoperable.

Technical References:

Lesson Plan GLP-OPS-L1100, GFIG-OPS-L1100, GGNS Technical Specifications LCO 3.8.4, 3.8.6, 3.8.7, and 3.5.1

References to be provided to applicants during exam:

GGNS Technical Specification 3.8.4

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5, 55.45.6	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation pump speed mismatch between loops. (CFR 41.5 / 45.6) (A2.04)	Tier #	2
	Group #	2
	K/A #	202002 A2.04
	Rating	3.2

Question 91

The plant is operating at 80% power when Reactor Recirc Pump 'A' unexpectedly downshifts to slow speed.

It's immediately determined that an I&C technician error was the cause of the downshift, and the problem has been corrected.

The reactor is now stable and operating in the Restricted Region of the Power-to-Flow Map.

Based on this, the CRS should enter Procedure:

A. 05-1-02-III-3 (Reduction in Recirc System Flow Rate) and upshift Reactor Recirc Pump 'A' to exit the Restricted Region.

B. 05-1-02-III-3 (Reduction in Recirc System Flow Rate) and monitor for power oscillations.

C. 04-1-01-B33-1 (Reactor Recirc System SOI) and upshift Reactor Recirc Pump 'A' to exit the Restricted Region.

D. EP-2 (RPV Control) and place the Mode Switch in SHUTDOWN to preclude excessive power oscillations.

Answer: B

Explanation:

A. This is incorrect because the referenced procedure contains a caution not to upshift the reactor recirc pump.

B. This is correct because the downshift of the reactor recirc pump requires entering this procedure. The procedure states to monitor power for power oscillations and place the MODE switch in SHUTDOWN if oscillations are observed.

C. This is incorrect because the crew should enter the abnormal procedure and the slow running recirc pump should not be upshifted.

D. This is incorrect unless there are power oscillations observed.

Technical References:

Procedure 05-1-02-III-3, Reduction in Recirc System Flow Rate

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to (a) predict the impacts of the Compressor trips (loss of air) (plant-specific) on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) (A2.08)	Tier #	2
	Group #	2
	K/A #	223001 K3.02
	Rating	3.1

Question 92

The plant is operating at rated power when Plant Air Compressor 'C' trips.

No other compressors have automatically started, and none can be manually started from the Control Room.

While implementing ONEP 05-1-02-V-9, Loss of Instrument Air, one control rod begins drifting into the core, and main condenser vacuum degrades to the point at which the main turbine vacuum trip is imminent.

Based on these conditions, the CRS should now enter:

A. ONEP 05-1-02-IV-1, Control Rod/Drive Malfunctions, ONLY, and scram the reactor based on the drifting control rod.

B. ONEP 05-1-02-V-8, Loss of Condenser Vacuum, ONLY, and perform a fast shutdown based on the imminent turbine trip.

C. ONEP 05-1-02-IV-1 <u>and</u> ONEP 05-1-02-V-8, then perform a fast shutdown based on the drifting control rod.

D. ONEP 05-1-02-IV-1 <u>and</u> ONEP 05-1-02-V-8, then scram the reactor based on the imminent turbine trip.

Answer: D

Explanation:

A. This is incorrect because the Loss of Vacuum ONEP must also be entered, and one drifting rod does not require a scram.

B. This is incorrect because the CRD Malfunctions ONEP must also be entered, and a scram is required rather than a fast shutdown.

C. This is incorrect because the drifting control rod does not require a scram.

D. This is correct because both ONEPs must be entered and a scram is required based ion the imminent turbine trip.

Technical References:

Off-Normal Event Procedure (ONEP) 05-1-02-V-9, Loss of Instrument Air , ONEP 05-1-02-IV-1, Control Rod/Drive Malfunctions, and ONEP 05-1-02-V-8, Loss of Condenser Vacuum, EN-OP-115, Conduct of Operations, ONEP for Reduction in Recirc System Flowrate

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Knowledge of new and spent fuel movement procedures. (CFR: 41.10 / 43.7 / 45.13) (2.1.42)	Tier #	2
	Group #	2
	K/A #	233000 G2.1.42
	Rating	3.4

Question 93

A Precaution/Limitation in IOI 03-1-01-5, "Refueling," states that the maximum allowed weight of a load traveling over irradiated fuel assemblies in the Spent Fuel or Upper Containment Storage Pool Racks is _____ pounds.

A. 1000

B. 1140

C. 1350

D. 1430

Answer: B

Explanation:

A. This is incorrect; the referenced procedure states in caution/limitation 2.8 the maximum weight traveling over the Spent Fuel or Upper Containment Storage Pool is 1140 pounds.

B. This is correct per the discussion above.

C. This is incorrect; the referenced procedure states in caution/limitation 2.8 the maximum weight traveling over the Spent Fuel or Upper Containment Storage Pool is 1140 pounds.

D. This is incorrect; the referenced procedure states in caution/limitation 2.8 the maximum weight traveling over the Spent Fuel or Upper Containment Storage Pool is 1140 pounds.

Technical References:

Integrated Operating Instruction 03-1-01-5, Refueling

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.7	

Examination Outline Cross-Reference	Level	SRO
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)	Tier #	3
	Group #	
	K/A #	G 2.1.5
	Rating	3.9

Question 94

With the reactor in Mode 4, Technical Specification 5.2.2, Unit Staff, requires which of the following, by position, as the minimum for unit staffing?

A. 2 non-licensed operators, 2 licensed reactor operators, 1 licensed senior reactor operators, and no health physicists.

B. 1 non-licensed operator, 1 licensed reactor operator, 1 licensed senior reactor operator, and 1 health physicists.

C. No non-licensed operators, 1 licensed reactor operator, 1 licensed senior reactor operator, and no health physicists.

D. No non-licensed operators, 1 licensed reactor operator, 1 licensed senior reactor operator, and 1 health physicists.

Answer: B

Explanation:

A. This is incorrect because with the reactor in Mode 4, there is fuel in the RPV. According to the referenced TS, a health physicist is required to be on-site.

B. According to the referenced TS, these are the minimum staffing requirements.

C. This is incorrect because with the reactor in Mode 4, there is fuel in the RPV. According to the referenced TS, a health physicist and one NLO are required to be on-site.

D. This is incorrect because with the reactor in Mode 4, there is fuel in the RPV. According to the referenced TS, an NLO is required to be on-site.

Technical References:

TS Technical Specification 5.2.2, Unit Staff

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)	Tier #	3
	Group #	
	K/A #	G 2.1.23
	Rating	4.4

Question 95

Following a LOCA with limited fuel damage, the Shift Technical Advisor reports containment hydrogen concentration has reached 3%.

Based on this;

A. EP-2, RPV Control, is exited and EP-3, Containment Control is continued.

B. EP-2, RPV Control, is exited, EP-5, RPV Flooding is entered, and EP-3, Containment Control, is continued.

C. EP-2, RPV Control, is continued, EP-3, Containment Control is exited, and the SAPs are entered.

D. All EPs are exited and all SAPs are entered.

Answer: D

Explanation:

A. This is incorrect because the EPs state if containment or drywell hydrogen concentration exceeds 2.9%, all the EPs are exited and all the SAGs are entered.

B. This is incorrect because the EPs state if containment or drywell hydrogen concentration exceeds 2.9%, all the EPs are exited and all the SAGs are entered

C. This is incorrect because the EPs state if containment or drywell hydrogen concentration exceeds 2.9%, all the EPs are exited and all the SAGs are entered

D. This is correct based on the discussion above.

Technical References:

EP-3, Containment Control, step H-1.

References to be provided to applicants during exam:

None

Learning Objective:

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
Ability to explain and apply system limits are precautions. (CFR: 41.10 / 43.2 / 45.12)	Tier #	3
	Group #	
	K/A #	G 2.1.32
	Rating	4.0

Question 96

According to Technical Specification 3.6.4.3, two Standby Gas Treatment trains are required to be OPERABLE during normal power operations.

One of the Technical Specification Bases for this limit is to mitigate the radiological consequences of a:

A. Rod drop accident low in the power range.

B. Fuel handling accident involving irradiated fuel.

C. Stuck open SRV while at power.

D. Power excursion resulting in greater than 3 percent fuel damage.

Answer: B

Explanation:

A. This is incorrect because the event would result in fuel damage and the release of fission products; SGTS would not mitigate the radiological release unless there is leakage of fission products into the secondary containment.

B. This is correct per TS Basis for TS 3.6.4.3. The basis is for a DBA LOCA and a Fuel handling accident involving irradiated fuel.

C. This is incorrect because the event would result in the release of reactor coolant into the suppression pool, SGT would not mitigate the radiological release.

D. This is incorrect because even though the event did result in fuel damage and the release of fission products, SGT would not mitigate the radiological release unless there is leakage of the fission products into the secondary containment.

Technical References:

Technical Specification Basis for TS 3.6.4.3, Standby Gas Treatment

References to be provided to applicants during exam:

None

Learning Objective:

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)	Tier #	3
	Group #	
	K/A #	G 2.2.25
	Rating	4.2

Question 97

One of the Technical Specification safety limits is the requirement to maintain the Minimum Critical Power Ratio (MCPR) greater than or equal to 1.08 for two-loop operation when greater than 25% power.

The Technical Specification bases for this limit is to ensure:

A. 99.9% of the fuel rods do not experience transition boiling.

B. the peak cladding temperature limit per 10 CFR 50.46 is not exceeded after a LOCA.

C. the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) is not exceeded.

D. that following a LOCA, decay heat does not exceed the design of the containment systems.

Answer: A

Explanation:

A. This is correct because Technical Specification bases state this is the basis for the MCPR safety limit.

B. This is incorrect because Technical Specification bases state this is the basis for the MCPR safety limit.

C. This is incorrect because Technical Specification bases state this is the basis for the MCPR safety limit.

D. This is incorrect because Technical Specification bases state this is the basis for the MCPR safety limit.

Technical References:

Technical Specification bases document for LCO 2.1.1, Reactor Core Safety Limits

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.	

Examination Outline Cross-Reference	Level	SRO
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)	Tier #	3
	Group #	
	K/A #	G 2.2.42
	Rating	4.6

Question 98

The on-coming Shift Manager is performing a control room walkdown prior to taking the shift, and notes the following:

- Reactor power is 80%
- Reactor Recirc loop 'A' flow is 35,000 gpm
- Reactor Recirc loop 'B' flow is 33,700 gpm
- One ADS valve is tagged out with a red DO NOT OPERATE tag
- Suppression pool level is 18' 5"
- Primary Containment air temperature is 89°F

The on-coming Shift Manager should verify an entry has been made into Technical Specification:

A. 3.4.1, Recirculation Loops Operating

B. 3.5.1, ECCS Operating

C. 3.6.1.5, Primary Containment Air Temperature

D. 3.6.2.2, Suppression Pool Water Level

Answer: B

Explanation:

A. This is incorrect because both loops are operating and they are within 5% of each other as required by this TS.

B. This is correct because all 8 ADS valves are required to be operable in Mode 1.

C. This is incorrect because 89 degrees is within the TS limit of 95 degrees and Administrative Limit of 90 degrees per CR 2001-1951.

D. This is incorrect because suppression pool level is within the TS required band.

Technical References:

3.6.1.5, Primary Containment Air Temperature, 3.4.1, Recirculation Loops Operating, 3.5.1, ECCS Operating, 3.6.2.2, Suppression Pool Water Level

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)	Tier #	3
	Group #	
	K/A #	G 2.3.11
	Rating	4.3

Question 99

According to procedure 01-S-08-11, Radioactive Discharge Controls, final release authorization of a Batch Liquid Radwaste Discharge is the responsibility of the:

A. CRS ONLY.

B. Shift Manager ONLY.

C. CRS and Chemistry Supervisor/Coordinator.

D. Shift Manager and Chemistry Supervisor/Coordinator.

Answer: B

Explanation:

A. Incorrect – The referenced procedure states this is the responsibility of the SM.

B. Correct – The referenced procedure states in section 2.1, and in step 6.4.2.d, that this is a SM responsibility. The final release signature is by the SM in section 4 of the batch release form.

C. Incorrect – The Chemistry Supervisor does have responsibilities for a batch release, but they do not include final release approval.

D. Incorrect – The Chemistry Supervisor does have responsibilities for a batch release, but they do not include final release approval.

Technical References:

Procedure 01-s-08-11, Radioactive Discharge Controls

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

Examination Outline Cross-Reference	Level	SRO
Knowledge of the lines of authority during implementation of the emergency plan. (CFR: 41.10 / 45.13)	Tier #	3
	Group #	
	K/A #	G 2.4.37
	Rating	4.1

Question 100

The site has entered a General Emergency and consideration is being given to evacuating 5 sectors of the 10 mile Emergency Planning Zone (EPZ).

The decision to order an evacuation of these sectors will be made by the:

A. Emergency Director

B. Off-Site Emergency Coordinator

C. State and Local Agencies

D. Radiation Emergency Manager

Answer: C

Explanation:

A. This is incorrect because the ED is responsible for actions taken on site but only makes recommendations for off-site actions such as PARs.

B. This is incorrect because the ED is responsible for actions taken on site but only makes recommendations for off-site actions such as PARs.

C. This is correct because the responsibility for deciding to order an evacuation of any EPZ is made by the State and Local Agencies.

D. This is incorrect because the REM is responsible for making recommendations to the ED.

Technical References:

10-S-01-1 and 10-S-01-12

References to be provided to applicants during exam:

None

Learning Objective: GLP-OPS-

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	